TECHNICAL PROGRESS REVIEWS

POWER REACTOR TECHNOLOGY

Prepared for U. S. ATOMIC ENERGY COMMISSION by GENERAL NUCLEAR ENGINEERING CORP.

DECEMBER 1958 VOLUME 2 NUMBER 1

TECHNICAL PROGRESS REVIEWS

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POWER REACTOR TECHNOLOGY

a review of recent developments *prepared by*

GENERAL NUCLEAR ENGINEERING CORP.

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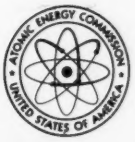

foreword

This quarterly review of reactor development has been prepared at the request of the Division of Information Services of the United States Atomic Energy Commission. Its purpose is to assist interested organizations in the task of keeping abreast of new results in reactor technology for civilian application.

The report is a concise discussion of selected phases of research and development for which there have been significant advances or a heightened interest in the past few months. It is not meant to be a comprehensive abstract of all material published during the quarter nor is it meant to be a treatise on any part of the subject. The intention is to cover the various areas of reactor development from the general viewpoint of the reactor designer rather than from the more detailed points of view of specialists in the individual areas. However, papers which are thought to be of particular significance or of particular usefulness in specialized fields will be mentioned in short notes. In the over-all plan of the report, it is intended that various subjects will be treated from time to time and will be brought up to date at that time.

Any interpretation of results which is given represents only the opinion of the editors of the report, who are General Nuclear Engineering Corporation personnel. Readers are urged to consult the original references in order to obtain all the background of the work reported and to obtain the interpretation of the results given by the original authors.

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contents

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PROGRESS ON SPECIFIC REACTOR TYPES

INTRODUCTION

As a result of the Second International Conference on the Peaceful Uses of Atomic Energy, a large amount of information has been published which describes the status of almost all power reactors currently in operation, under construction, or in advanced stages of design. In addition to the more than 2250 papers submitted by the attending nations, a special presentation set of thirteen volumes on atomic energy was prepared by the United States for the Conference.

The pressurized light-water reactors are covered by a book on the Shippingport reactor[1] and a number of Conference papers on other reactors[1-6] of the type presently under construction.

A book[7] has also been devoted to the boiling-water reactors, and Conference papers cover not only boiling power reactor designs[8,9] but also fundamental questions of boiling reactor technology.[10-12]

The British and French gas-cooled reactors are summarized in a series of Conference papers,[13-25] and work in the United States on this reactor type has been covered in one of the presentation volumes.[26]

The United States work on the sodium graphite reactor is the subject of one of the presentation volumes,[27] and a Conference paper covers operating experience in the Sodium Reactor Experiment (SRE).[28] The USSR has announced work on a sodium graphite reactor of 50-Mw (electrical) output,[29] while the British have stated that their work on sodium graphite has been discontinued, at least for the present.

In the field of fast reactors, the work in the United States is covered rather extensively in the presentation volume on solid fuel reactors,[26] and several Conference papers also cover designs of specific reactors.[30-32]

Conference papers also bring up to date the design information on the British Dounreay reactor[33] and describe the Russian program,[34] which includes a 5-Mw (thermal) reactor which is completed, a 50-Mw (electrical) reactor which is under design, and a 250-Mw (electrical) station which is in the planning stage.

The fluid fuel reactors are covered in one of the presentation volumes,[35] and recent progress is described in a number of Conference papers.[36-41]

The aqueous homogeneous[42] reactors and the liquid metal fuel reactors[43] have also been reviewed recently in this publication.

Of central interest in the field of organic-cooled and -moderated reactors is the Organic Moderated Reactor Experiment (OMRE) which has now been in operation for a few months. This experiment is described in one of the presentation volumes[26] and in a Conference paper.[44]

The D_2O-cooled and -moderated reactor appears to command geographically the most widespread interest, although no power reactor of this type has yet been built in any size. The Canadian work on this reactor is the most extensive; it was described at the Conference,[45,46] and many Canadian reports have been issued covering the technology of this reactor type. Work in the United States is summarized in the volume on solid fuel reactors[26] and by a Conference paper.[47]

Designs of reactors being planned in Norway[48] and Switzerland[49] were also described at the Conference.

In Table I-1, reactor design and construction projects are listed according to reactor type and country, and appropriate references to the published information are given.

Of the reactor types listed above, there are two which are represented by enough operating reactors and reactor designs to be considered for the present as more or less standardized, with characteristics which will vary from re-

Table I-1 — POWER REACTOR PROJECTS AND DESIGN STUDIES

Reactor	Electric power, Mw: Per reactor	Per station	Status	Country	Reference
		Nonboiling H_2O			
Shippingport	60	60	Operating	USA	1, 2
Yankee	134	134	Constructing	USA	3, 52
Consolidated Edison (Indian Point)	275	275	Constructing	USA	4, 52
APPR-1	2	2	Operating	USA	6, 52–54
Savannah Ship	20,000 shaft H.P.	20,000 shaft H.P.	Constructing	USA	55–57
Voronezh-PWR	210	420	Constructing	USSR	5
Leningrad-PWR	210	420	Planned	USSR	29
Soviet-PWR-Mobile	2	2	Constructing	USSR	29
Ice Breaker "Lenin"	22,000 shaft H.P.	66,000 shaft H.P.	Constructing	USSR	58
Emigrant Ship	180 (thermal)	180 (thermal)	Design study	Japan	59
Submarine Tanker	180 (thermal)	180 (thermal)	Design study	Japan	60
		Boiling H_2O			
BORAX-IV	3.5 (max.)	3.5 (max.)	Operating	USA	7
EBWR	5	5	Operating	USA	7, 10
VBWR	5	5	Operating	USA	7, 11
Dresden	192	192	Constructing	USA	7, 8
Elk River	22	22	Planned	USA	55
Northern States	62	62	Planned	USA	52, 55
Pacific Gas and Electric	50	50	Planned	USA	55
Soviet BWR	50	50	Constructing	USSR	29
Belgonucleaire IC-BWR	129	129	Design study	Belgium	9, 61
		Graphite-Moderated, Gas-Cooled			
Calder A	46	92	Operating	UK	62, 63
Calder B	46	92	Operating	UK	62, 63
Chapelcross	46	182	Constructing	UK	62, 63
Berkeley	166	332	Constructing	UK	62, 63
Bradwell	176	352	Constructing	UK	62, 63
Hunterston	167	344	Constructing	UK	62, 63
Hinkley Point	313	626	Constructing	UK	62, 63
UK-AGR	28	28	Planned	UK	15
UK-HTGR	10 (thermal)		Planned	UK	19
ORNL-GCR	252	252	Design study	USA	26, 64, 65
K-ACF-GCR	253	253	Design study	USA	26, 64, 66
G-1	5	5	Operating	France	22
G-2	30	30	Constructing	France	22, 23
G-3	30	30	Constructing	France	22, 23

Table I-1 — (Continued)

Reactor	Electric power, Mw: Per reactor	Per station	Status	Country	Reference
		Fast Reactors			
EBR-II	20	20	Constructing	USA	26, 32, 67
Enrico Fermi	104	104	Constructing	USA	26, 30, 31, 52, 67
Dounreay	15	15	Constructing	UK	33, 67
BR-5	5 (thermal)	5 (thermal)	Operating	USSR	34
BN-50	50	50	Planned	USSR	34
BN-250	250	250	Planned	USSR	34
		Sodium-Graphite			
SRE	6	6	Operating	USA	27, 28
Hallam-SGR	75	75	Planned	USA	52
Soviet-SGR	50	50	Constructing	USSR	29
		Graphite-Moderated, H_2O-Cooled			
APS-1 Obninsk	5	5	Operating	USSR	50
Soviet-Ural	100	400	Constructing	USSR	29, 50
Soviet-Siberia	100	600	Operating	USSR	68
		Organic-Moderated			
OMRE	15 (thermal max.)		Operating	USA	26, 44, 69
Piqua	12	12	Planned	USA	52
		Homogeneous Aqueous			
HRE-2	5 (thermal)		Operating	USA	35, 37, 42
Pennsylvania Advanced Reactor	150 (max.)	150 (max.)	Planned	USA	35, 55
Czech-H_2O	10	10	Planned	Czechoslovakia	36
Soviet-D_2O Boiling	35 (thermal)		Constructing	USSR	29
Dutch-H_2O	0.25 (thermal)		Constructing	Netherlands	70, 71
		D_2O-Moderated and Cooled			
NPD	20	20	Constructing	Canada	45, 46
CANDU	200	200	Design study	Canada	
Carolinas-Virginia	17	17	Planned	USA	55
Sulzer	30 (thermal)		Design study	Switzerland	49
Halden	10 (thermal)		Constructing	Norway	48
		D_2O-Moderated, Gas-Cooled			
DM-GCR	50	50	Planned	USA	26
Czech-GCR	150	150	Constructing	Czechoslovakia	29
		D_2O-Moderated, Sodium-Cooled			
Chugach	10	10	Planned	USA	52

actor to reactor but over a fairly well-defined range. These are the pressurized H_2O reactors (boiling and nonboiling) and the gas-cooled natural-uranium graphite-moderated reactors. The information presented by the USSR at the International Conference has introduced a third type, the water-cooled graphite-moderated reactor, which promises to be an important one. This type was described at the 1955 Conference as that of the First Atomic Power Station in the USSR, but the direction in which the concept would develop was not at that time apparent outside the USSR; and the general concept covers a wide range of possible reactors, as exemplified (1) by the dual-purpose reactor which recently went into operation in Siberia, with relatively low-temperature performance, and (2) by the enriched-uranium superheating reactor[50] currently under construction, which promises quite high steam temperatures.

It is perhaps reasonable to say that these three reactor types—H_2O-cooled and -moderated, gas-cooled graphite-moderated, and water-cooled graphite-moderated—have achieved economic acceptance. This is not intended to imply that they are competitive with fossil-fuel power plants; rather they have been built in large sizes, and their performance has been encouraging enough to justify the commitment of large sums of money for the construction of further large-scale plants of this type. It is believed that some further discussion of these types, beyond the published descriptions of reactor designs, is worth while. The remainder of this review will be devoted to such a discussion. Its objective will be to characterize the reactors of the three classes, to compare designs of different reactors within the classes for the purpose of arriving at generalizations, and to discuss the trend of future development.

In Table I-2 are listed some of the general characteristics of a number of specific large reactor designs or design studies which will have a bearing on the discussion. It is not intended that the performance numbers listed in the table be compared directly for the various reactors since such comparisons between detailed designs and design studies can be misleading; the intention is simply to collect in one place the general characteristics of a number of reactors which will be referred to rather frequently in later sections.

Only a few generalizations can be made with regard to the relative promise of the three reactor types. Of the single-purpose power reactors, it can probably be said that the natural-uranium gas-cooled type has the lowest fuel cost. The type suffers, however, from high capital cost, and substantial improvements in this characteristic appear to require the use of enriched fuel and more expensive fuel-element constructions. In the area of capital costs, engineering judgment suggests that it will be hard to find reactor types of potentially lower cost than the water-cooled and -moderated types. These reactors are handicapped, however, by their relatively low-temperature limitations. In Fig. 1 the coolant outlet temperatures and the thermal efficiencies have been plotted for a number of reactors of various types which have reached a fairly advanced stage of design or which are known to have had the benefit of rather careful design studies. The values are also tabulated in Table I-3. Of particular interest is the graphite, water-

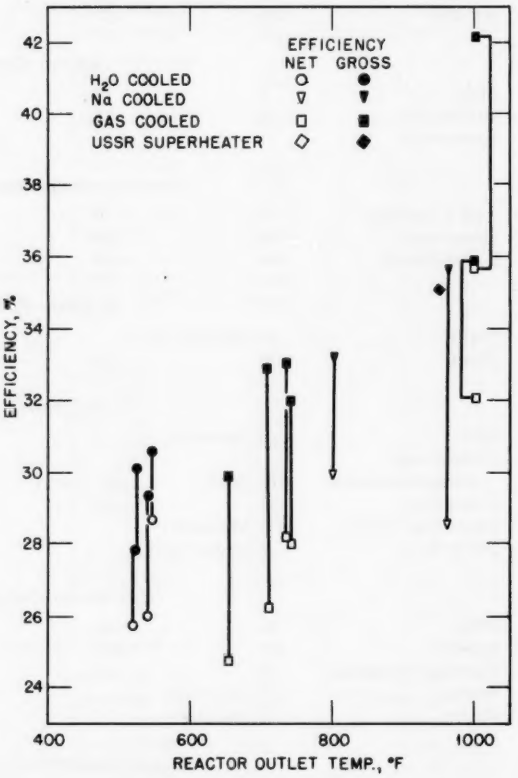

Figure 1—Outlet coolant temperatures and quoted efficiencies for several reactors (see Table I-3).

Table I.2—CHARACTERISTICS OF SOME POWER REACTOR DESIGNS AND DESIGN STUDIES

Type	Boiling Water Reactors			
Reactor	Dresden[7,8,72]		Belgonucleaire[9,61]	
Location of station	Morris, Ill.		Design study	
Reactor per station	1		1	
Power per reactor				
Thermal, Mw	626		241.9 + 92.8 from superheater	
Electrical, Mw (g)	192		128.5 (with superheater)	
Electrical, Mw (net)	180		124.2 (with superheater)	
Moderator	H_2O		H_2O	
Coolant	H_2O		H_2O	
Inlet temperature, °F	504		522	
Outlet temperature, °F	546		545	
Pressure, psia	1015		1000	
Fuel	UO_2		UO_2	
Shape	Rods		Rods	
Enrichment, %	1.5		2.5	
Weight, tons*	65.8 (UO_2)		23.53 (UO_2)	
Clad	Zr-2		S.S.	
Lifetime, Mwd/ton	10,000		10,000	
Conversion ratio	ICR = 0.7		ICR = 0.56	
Heat flux				
Maximum, Btu/(hr)(sq ft)	277,000†		365,000	
Average, Btu/(hr)(sq ft)	94,660		79,300	
Ratio	2.93		4.50	
Steam system	Dual pressure		Heat exchanger	Superheater
Pressure, psia	1015	510	660	615
Temperature, °F	546	469	497	1049

Type	Nonboiling Pressurized Water Reactors				
Reactor	Russian[5]	Yankee[3,52]		Consolidated Edison[4,52]	
Location of station	Voronezh, Russia	Rowe, Mass.		Indian Point, N. Y.	
Reactor per station	2	1		1	
Power per reactor		1st core	2nd core	1st core[52]	2nd core[4]
Thermal, Mw	760	392	482	500	585
Electrical, Mw (g)	210		145		163 reactor 275 reactor + S.H.
Electrical, Mw (net)		110	134		151 (est.)
Moderator	H_2O	H_2O	H_2O	H_2O	H_2O
Coolant	H_2O	H_2O	H_2O	H_2O	H_2O
Inlet temperature, °F	482	499	495	481.5	485
Outlet temperature, °F	527	529	532	510	517
Pressure, psia	1470	2000		1515	1515
Fuel	UO_2	UO_2		ThO_2-UO_2	ThO_2-UO_2
Shape	Rods	Rods		Rods	Rods
Enrichment, %	1.5	3.0		~4.2 = $\frac{U^{235}\text{ wt.}}{(UO_2 + ThO_2)\text{ wt.}}$	
Weight, tons*	44 (UO_2)	25.1 (UO_2)		19.2 (ThO_2) 1.1 (UO_2)	21.1 (ThO_2) 1.1 (UO_2)
Clad	Zr-Nb alloy	S.S.		S.S.	
Lifetime, Mwd/ton	8300	6550		18,000	
Conversion ratio	~0.75	ICR = 0.733		0.5	
Heat flux					
Maximum, Btu/(hr)(sq ft)	443,000	450,000		450,000	
Average, Btu/(hr)(sq ft)	80,000	87,000		112,500	
Ratio	5.54	5.18		4.01	
Steam system		Sat. steam			Sat. S.H.
Pressure, psia	426	540			405‡ 370‡
Temperature, °F	461‡	475			449‡ 1000‡

*Ton = 2000 pounds. †Max. heat flux at 125% rated power = 346,000 Btu/(hr)(sq ft). ‡At turbine inlet.

Table I-2 — (Continued)

Type	Graphite-moderated, Gas-cooled Reactors	
Reactor	Bradwell[62,64]	Hinkley Pt.[62,64]
Location of station	Essex, England	Somerset, England
Reactor per station	2	2
Power per reactor		
Thermal, Mw	531	950
Electrical, Mw (g)	176	313
Electrical, Mw (net)	150	250
Moderator	Graphite	Graphite
Coolant	CO_2	CO_2
Inlet temperature, °F	356	356
Outlet temperature, °F	734	707
Pressure, psia	147	~200
Fuel	U metal	U metal
Shape	Cylinders	Cylinders
Enrichment, %	Natural	Natural
Weight, tons*	261	414
Clad	Magnox A-12	Magnox A-12
Lifetime, Mwd/ton	~3000 (max.)	~3000 (max.)
Conversion ratio	~0.83	~0.85
Heat flux		
Maximum, Btu/(hr)(sq ft)	~159,000†	~139,000†
Average, Btu/(hr)(sq ft)	~97,000	~110,000
Ratio	1.64	1.25
Steam system	H.P. L.P.	H.P. L.P.
Pressure, psia	770 210	655 186
Temperature, °F	707 707	686 660
Reactor	ORNL-Optimum[26,64,65]	K-ACF-Optimum[26,64,66]
Location of station	Design study	Design study
Reactor per station	One	One
Power per reactor		
Thermal, Mw	700	600
Electrical, Mw (g)	252	253
Electrical, Mw (net)	225	215
Moderator	Graphite	Graphite
Coolant	Helium	CO_2
Inlet temperature, °F	450	473
Outlet temperature, °F	1000	1000
Pressure, psia	300	370
Fuel	UO_2	UO_2
Shape	Rods	Rods
Enrichment, %	2.0	2.5
Weight, tons*	170 (UO_2)	96 (UO_2)
Clad	S.S.	S.S.
Lifetime, Mwd/ton	7,350	10,000
Conversion ratio		ICR = 0.68
Heat flux		
Maximum, Btu/(hr)(sq ft)	96,000	
Average, Btu/(hr)(sq ft)	56,000	~85,000
Ratio	1.71	
Steam system		H.P. L.P.
Pressure, psia	950	2430 780
Temperature, °F	950	950 950

*Ton = 2000 pounds.
†Averaged over the maximum channel.

Table I-2—(Continued)

Type	Graphite-moderated, H_2O-cooled Reactors		
Reactor	USSR-Superheater[50]		USSR-Siberian[68]
Location of station	Beloyerski Urals, Russia		Siberia, Russia
Reactor per station	4		6 (?)
Power per reactor			
Thermal, Mw	285		
Electrical, Mw (g)	100		100
Moderator	Graphite		Graphite
	Boiling channel:	Superheat channel:	
Coolant	H_2O	Steam	Water
Inlet temperature, °F	572	607	365
Outlet temperature, °F	655	950	428
Pressure, psia	2280	1620	
Fuel	U–12 wt.% Mg		Uranium
Shape	Tubular		
Enrichment, %	1.3		Natural
Weight, tons*	90 (U)		
Clad	S.S.		Aluminum alloy
Lifetime, Mwd/ton	2300		
Conversion ratio	0.65		
Heat flux			
Maximum, Btu/(hr)(sq ft)	194,000	134,000	
Average, Btu/(hr)(sq ft)			
Ratio			
Steam system			
Pressure, psia	1320		<163
Temperature, °F	932		365°F superheated

*Ton = 2000 pounds.

Table I-3—COOLANT OUTLET TEMPERATURES AND THERMAL EFFICIENCIES FOR REACTOR DESIGNS WITH VARIOUS COOLANTS

	Reactor coolant	Reactor coolant outlet temp., °F	Gross efficiency, %	Net efficiency, %
Shippingport[1,2]	Nonboiling H_2O	538	30.2	26.7
Yankee[3,52] (2nd core)	Nonboiling H_2O	532	30.1	27.9
Consolidated Edison[4] (2nd core)	Nonboiling H_2O	517	27.9	25.8 (est.)
Dresden[7,8,72]	Boiling H_2O	546	30.7	28.8
USSR Superheating Reactor[50]	Boiling H_2O and steam	950	35.1	
Enrico Fermi Atomic Power Plant (Refs. 26, 30, 31, 52, and 67)	Sodium	800	33.3	30.0
Sodium Reactor Experiment[27,28]	Sodium	960	35.7	28.6
Berkeley[62,64]	Gas	653	29.9	24.8
Hinkley Point[62,64]	Gas	707	33.0	26.3
Bradwell[62,64]	Gas	734	33.1	28.2
Hunterston[62,64]	Gas	736	32.1	28.0
Kaiser Gas-Cooled (enriched)[26,64,66]	Gas	1000	42.2	35.8
ORNL Gas-Cooled (enriched)[64,65]	Gas	1000	36.0	32.1

cooled, superheated reactor. This reactor type has not had enough study in the United States to make possible estimates as to its place in the power-cost picture, but its estimated high-temperature performance is of interest as an indication of what may be accomplished with water (and steam) cooling.

The three reactor types are discussed separately in the following sections. As a supplement to this discussion, the reader is referred to the previous issue of *Power Reactor Technology*,[51] which summarizes the status of fuel-element development for several reactor types.

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In the field of reactors cooled and moderated by H_2O, there is a division into two classes of reactors: those which employ a significant amount of boiling in the coolant water and those which do not. In each of these classes the further division, between those reactors which use zirconium alloys for fuel jacketing and those which use steel as jacket material, represents significant differences in reactor performance. Despite these differences, there are many fundamental similarities which characterize water-moderated reactors. These characteristics are determined basically by the nuclear and thermal characteristics of H_2O. The reactors all employ lattices of relatively low moderator to fuel ratio. They all require enriched fuel for operation; they have relatively high excess reactivity; and, in existing designs, they display relatively low conversion ratio, although this characteristic is not necessarily an inherent one. All existing designs are of the pressure vessel type. Because of the high ratio of fuel to moderator, it is possible to design for relatively high total power output in a pressure vessel of reasonable diameter. Most of the discussion in the following sections will apply to the class of water-cooled and -moderated reactors as a whole; the characteristic differences between boiling and non-boiling reactors, and between those employing zirconium jackets and those employing steel jackets, will be pointed out at appropriate points.

Reactor Core Materials

As fuel material, uranium dioxide (or a mixture of uranium and thorium dioxides) has been almost universally adopted. The use of oxide rather than metal results from two major considerations. Long fuel life appears to be necessary for economic operation of a reactor using enriched fuel in order that the fuel reprocessing costs may be distributed over a large amount of produced energy. It does not presently seem possible to achieve very long lifetime with metallic fuels except perhaps with alloys which employ rather high percentages of high-cross-section alloying materials. However, experiments indicate the possibility of achieving quite long fuel life when oxide is used. The second consideration in favor of the oxide is the rather violent chemical reaction of unalloyed uranium metal with hot water. The main deficiency of oxide as a fuel-element material is its rather low thermal conductivity. During the several years of development of oxide as a reactor fuel, estimates of the effective value of thermal conductivity of the oxide in an operating fuel element have tended to decrease.*

For fuel-element jackets there are two popular alternatives: zirconium alloy, which gives the best neutron economy but at present is the more expensive, and stainless steel. In the United States the zirconium alloy which has been developed and proved for this service is Zircaloy-2. In the USSR, a zirconium-niobium alloy (2.5 per cent niobium) has been developed which is said to have the advantages[2] of easier production and higher strength at temperatures in the 300°C range. Although economic considerations enter into the choice between zirconium and steel alloys for fuel jackets, it is probable that the major consideration which has favored the use of steel is the desire to minimize the probability of violent chemical re-

* The most recent investigations have been reported by WAPD.[1] Pellets of UO_2, 0.3535 in. in diameter, were loaded into a stainless-steel capsule with a diametral clearance of 0.0035 in. Heat output of the capsule was determined from measured temperature gradients in the thick stainless-steel capsule. Centerline temperature of the UO_2 was measured with a platinum, platinum-rhodium thermocouple. Irradiations were carried out in the MTR at various power levels. Total nvt was only 0.7×10^{20}. The "effective" thermal conductivity appears to be 1.0 ± 0.2 Btu/(hr)(ft)(°F) between 750 and 1100°F, calculated on the assumption that the 3.5-mil diametral gap (filled with helium) is preserved. On the assumption that the UO_2 and helium are smeared out over the full diameter of the capsule, the effective thermal conductivity is calculated to be 0.85 ± 0.7 Btu/(hr)(sq ft)(°F).

actions between metal and water in the event of accidental melting of the fuel-element jackets.[3]

With respect to its normal performance as a fuel-jacket material, Zircaloy has been proved more thoroughly than stainless steel, but neither material is expected to have serious troubles. There has, however, been some evidence of radioactivity problems stemming from the corrosion products of stainless-steel jackets,[4,5] and after very long exposure to hot water, Zircaloy has been observed to undergo some increase in corrosion rate* (the so-called "breakaway corrosion"). As an alternative metal for fuel jackets, combining the features of low cost and low neutron absorption, aluminum alloys have been under development for some time.[7] Although the results have been encouraging, aluminum is currently being considered seriously for only one commercial H_2O-moderated reactor, the Northern States Nuclear Power Plant.[8] The long fuel lifetime necessary in water-moderated reactors (because of the economics of the enriched fuel cycle) sets a difficult goal for aluminum-jacketed fuel elements, and it is not surprising that the first use of aluminum jackets in power reactors is in a natural-uranium reactor of low thermal performance, the recently announced water-cooled graphite-moderated reactor of 100-Mw electrical output, located in Siberia.[9]

Currently fabrication costs for water-reactor fuel elements are high, even when steel is used as the jacket material. Factors contributing to the high cost are: close manufacturing tolerances resulting from rather high ratios of fuel-element surface to volume and the lack of a good thermal bond between fuel and jacket; the necessity for careful design and construction to avoid neutron flux peaks and to minimize hot channel factors; careful materials and construction controls to avoid corrosion susceptibility; and fabrication and inspection costs associated with fuel subassemblies which are made up of many small elements. Because of the high fabrication costs and the cost of reprocessing partially enriched fuel, high fuel exposures are necessary to achieve acceptably low specific fuel costs. Water reactors are characteristically designed for average fuel lifetimes of about 10,000 Mwd/ton; the maximum burn-up experienced by a fuel element may be higher, depending on the maximum to average ratio for the power distribution in the reactor and the degree to which fuel-element positions are shifted during the life of a given element. Irradiation tests in the United States have indicated that fuel lifetimes in this range are obtainable, although actual reactor experience will be required to establish such a result on a statistical scale. Results of tests in the USSR have now been reported which tend to confirm the test results in the United States. The Russian results on irradiation of complete subassemblies of fuel elements are particularly interesting.[2]

Because of the closely packed structure of the lattices in H_2O-moderated reactors, the adverse effects of neutron flux peaking in water holes, and the rather high excess reactivity requirement, there is a premium on control-rod materials which will give high neutron absorption per unit volume. The problem of control-rod design is intensified by the rather high levels of power generation in water-reactor control rods and by the necessity for resistance to aqueous corrosion. Most water lattices have relatively high ratios of epithermal to thermal neutron fluxes; consequently materials having high absorption in the epithermal region are attractive for control-rod use. Hafnium is a particularly attractive material which has been used in the Shippingport reactor and for some of the rods of EBWR. It has strong neutron absorption resonances, it is a good structural material, and it has corrosion resistance comparable to that of zirconium. Its thermal cross section is somewhat lower than the optimum, and it must therefore be used in relatively large thicknesses (of the order $\frac{1}{4}$ in.) for maximum effectiveness. Its disadvantages are its high cost and low availability. Boron steel[10] has been used successfully for control rods in water reactors. Its life may be shorter than is desirable if it is used in high flux regions because of the relatively low concentrations of boron which can be used and possibly because of the effects of alpha-particle

* The following summary is from Ref. 6. At temperatures below 600°F breakaway, corrosion does not occur for thousands of hours. The post-breakaway corrosion rate for unalloyed zirconium is approximately 100 mg/(cm^2)(mo). This would probably prevent the use of unalloyed zirconium as a long-term structural material. Zircaloy-2, which has a higher pre-breakaway corrosion rate than unalloyed high-purity zirconium, has a much smaller post-breakaway corrosion rate; approximately 7 mg/(cm^2)(mo).

damage. A cadmium-silver-indium rod is proposed for the Yankee reactor; it is expected to be corrosion-resistant. Although cheaper than hafnium, the materials for such a rod are still relatively expensive. Considerable development has been done on control rods utilizing the rare earths. Recent progress on this approach, and on the general problem of control rods for water reactors, is summarized in the papers of Ref. 11.

Nuclear Performance

Naturally the nuclear characteristics of H_2O-moderated reactors are to a large extent determined by the nuclear properties of H_2O. Nevertheless, in typical reactors of this type, the fuel jackets and structural material make up a sizeable fraction of the total and affect the nuclear performance importantly. In Table II-1 the volume ratios of materials in the core proper of the Yankee reactor[12] are listed. About 15 per cent of the core volume is occupied by structure and fuel-element jackets. This relatively high proportion of nonmoderating, nonfuel material complicates the discussion of the reactor physics.

Table II-1 VOLUME PERCENTAGES OF PRINCIPAL MATERIALS IN THE YANKEE ATOMIC ELECTRIC REACTOR[12]

Material	Density, g/cm^3	Volume, in.^{3}	Volume, %
UO_2	10.07	138,000	35.0
H_2O	1	198,800	50.5
Zirconium	6.56	12,600	3.2
Stainless steel (type 304)	8.03	44,600	11.3

There are two aspects of the nuclear performance that have important effects on the cost of nuclear power: the fuel enrichment and the conversion ratio. To a limited extent these two characteristics can be adjusted by reactor design: increases in enrichment make possible increases in conversion ratio. The design parameter which must be adjusted to preserve the neutron balance when such changes are made is the fuel to moderator ratio.

For moderators, the square root of the moderating ratio

$$\sqrt{\frac{\xi\Sigma_s}{\overline{\Sigma}_a}}$$

may be regarded as a figure of merit that determines the lowest fuel enrichment at which a large reactor employing the moderator will be critical when the fuel to moderator ratio is adjusted to the optimum value. The scattering cross section to be used in the expression for moderating ratio is that in the energy range of the U^{238} resonances, whereas the appropriate value for $\overline{\Sigma}_a$ is the average absorption cross section in the thermal energy region. This figure of merit for H_2O is about 8.4. Experiments have shown that a simple lattice of uranium metal and H_2O of optimum configuration can achieve a value of k_∞, the infinite multiplication constant, just less than unity. Thus a reactor moderated by H_2O cannot be made to operate with natural uranium. If uranium oxide is used instead of uranium metal, the situation is somewhat worse because the lower density of the oxide increases the surface to mass ratio for a given fuel geometry and thereby increases the effective resonance absorption integral (per atom) of the U^{238} in the fuel. If material which has a very low moderating power, such as zirconium or steel, is added to the fuel–H_2O mixture, the value of the quantity

$$\sqrt{\frac{\xi\Sigma_s}{\overline{\Sigma}_a}}$$

for the combination of H_2O and added material becomes less. In Table II-2 the quantity is given for pure H_2O and for a combination of H_2O plus fuel-jacket material. The case considered corresponds to fuel cylinders with a diameter

Table II-2 VALUES OF (MODERATING RATIO)$^{1/2}$ FOR MODERATORS AND MODERATOR–FUEL-JACKET MIXTURES

Material	$\sqrt{\xi\Sigma_s/\Sigma_a}$
H_2O	8.39
H_2O plus Zircaloy-2 jackets	8.25
H_2O plus stainless-steel jackets	5.94
Graphite (σ_a (kT) = 4.0 mb)	14.5

Note: For the cases involving fuel jackets, it is assumed that the jackets are 0.02 in. thick, are on fuel cylinders of diameter 0.40 in., and that the ratio (volume H_2O)/(volume fuel) = 2.5.

of 0.40 in., jacket thickness of 0.02 in., and H_2O to fuel volume ratio of 2.5. Also included in the table is the quantity

$$\sqrt{\frac{\xi\Sigma_s}{\bar{\Sigma}_a}}$$

for graphite. The latter value can serve as a bench mark since natural-uranium graphite reactors may be considered to have barely enough reactivity for practical power-reactor operation. Thus it is evident from the table that H_2O reactors inherently require enriched fuel and that the degree to which enrichment is required may be influenced strongly by the nature of the fuel jackets and other reactor structure. Later considerations of current water-moderated reactor designs will show that such reactors when fueled with UO_2 in zirconium jackets may require enrichments in the range about 1.5 per cent and when jacketed with stainless steel may have enrichments in the range about 3 per cent. Although these numbers are typical, they are by no means essential characteristics of the reactor type, as will be evident from the discussion below.

Although H_2O-moderated reactors are inherently limited to the use of enriched uranium, they are not inherently limited to low conversion ratio. To achieve high conversion ratio, two conditions must be met: the fraction of neutrons lost to parasitic absorption and by leakage from the reactor must be made small; and the neutrons in excess of those which are absorbed parasitically, those which leak, and those which are used for fission must be captured by a fertile material (U^{238} or thorium). Neutron leakage is typically low from H_2O-moderated reactors of central station size because of the short migration length of neutrons in a water–fuel mixture. The fractional parasitic absorption depends on the ratio of the macroscopic absorption cross sections of parasitic absorbers (i.e., all absorbers except fuel and fertile material) to the macroscopic absorption cross section of fissionable isotope. The fractional parasitic absorption can be decreased in any given situation by increasing the concentration of fissionable isotope (i.e., by increasing the enrichment). If the potential conversion ratio (i.e., the number of neutrons available for conversion per U^{235} atom destroyed) is designated by R_{pot}, then*

$$R_{pot} = \frac{\eta_{25}\epsilon}{1 + \tau B^2} - 1 - R_{f_p} - R_p - \gamma L^2 B^2 \quad (1)$$

In this expression the first term gives the number of neutrons, per U^{235} atom destroyed, which slow down in the reactor to the energy of the U^{238} absorption resonances. η_{25} is the average number of neutrons produced per neutron absorbed in U^{235}, ϵ is the fast fission factor, and $1 + \tau B^2$ accounts for the leakage of neutrons from the reactor during the slowing down period. For large reactors, $1 + \tau B^2$ is not far different from unity. The last term of the equation accounts for the thermal neutrons lost by leakage. The factor γ is the ratio of the total number of thermal neutrons absorbed to the number of thermal neutrons absorbed in U^{235}. It will always lie between one and two; in a water-moderated reactor its influence is small because the product L^2B^2 is quite small. The term R_{f_p} is the ratio of the macroscopic thermal absorption cross section of the short-lived fission products (xenon and samarium) to the macroscopic thermal absorption cross section of U^{235}. The term R_p is the ratio of the cross section of all other parasitic absorbers to the cross section of U^{235}. Obviously, R_p can be decreased at will by increasing the fuel enrichment.

To illustrate the effect of jacket material and enrichment on conversion ratio, Fig. 2 has been plotted for the case of the fuel-element-jacket–moderator combination previously considered in Table II-2. For the three cases—no jacket, Zircaloy-2 jacket, and stainless-steel jacket—the ratio $\bar{\Sigma}_a$ (H_2O + jacket)/$\bar{\Sigma}_a$ (U^{235}) is plotted as a function of enrichment. If there were no other parasitic absorbing material, the ratio plotted would be equal to the ratio R_p of Eq. 1. At an enrichment of 1.5 per cent, with the Zircaloy-2 jacket, there is a loss of 0.2 in the potential conversion ratio because of absorption in H_2O and jacket material; at the same enrichment, with the stainless-steel jacket, the loss

*In this discussion it is assumed that the reactor can be treated as a purely thermal reactor by the two-group approximation. More complex treatments are often appropriate, but they do not change the general trend of the argument.

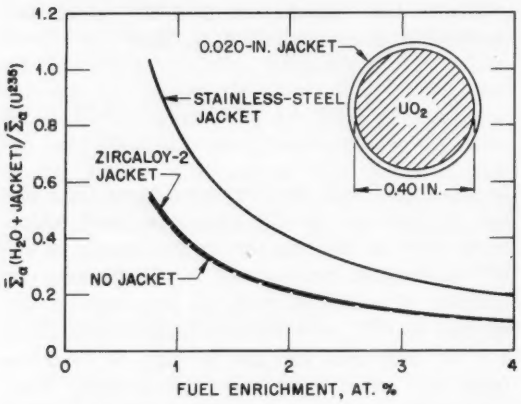

Figure 2—Effect of enrichment on neutron loss to H_2O and fuel jackets. Volume H_2O/volume UO_2 = 2.5; UO_2 density = 10.53 g/cm^3. Jacket and H_2O cross sections increased by 10 and 20 per cent, respectively, to allow for disadvantage factor.

would be 0.38. If, however, the enrichment is increased to about 2.8 per cent, the neutron loss for the steel jacket case can be reduced to about 0.2.

In the enrichment range above about 1.5 per cent, the net fuel cost (in mills/kw-hr) is not highly sensitive to enrichment provided specific power is high enough that inventory charges do not make an important contribution to the net fuel cost, and provided the fissionable isotope produced by conversion will command a price comparable to that of the initial fissionable isotope. For example, according to the AEC price schedule for partially enriched uranium, the cost of enriching from 1 to 2 per cent is $144.25 per kilogram of product, while the cost of enriching from 2 to 3 per cent is not much greater, $155 per kilogram of product. Consequently, in this range of enrichment, increases of enrichment may be used without large economic penalties to achieve the conversion ratio necessary for economically long fuel life or to simplify other problems of reactor design.

To make use of the neutrons which are available for conversion, the composition of the reactor must be so adjusted that they are captured in U^{238} (or thorium) in the reactor core. Limits are placed on the adjustment of composition by factors other than reactor physics, and it is therefore not necessarily always true that the designer can select the core composition which he would desire to obtain optimum physics performance. The actual conversion ratio R in the core is given by*

$$R = \frac{\eta_{25}\epsilon\,(1-p)}{1+\tau B^2} + \left(\frac{\Sigma_a\,(U^{238})}{\Sigma_a\,(U^{235})}\right)_{\text{thermal}} \qquad (2)$$

The first term of this equation is the same as that of Eq. 1 except that the multiplier (1 – p) has been added. p is the resonance escape probability for U^{238}, and the term now represents the number of neutrons absorbed epithermally in U^{238} per U^{235} atom destroyed. The second term, which is the ratio of the macroscopic thermal absorption cross section of U^{238} to the macroscopic thermal absorption cross section of U^{235}, gives the number of neutrons absorbed in U^{238} thermally per U^{235} atom destroyed. If the actual conversion ratio as given by Eq. 2 is equal to the potential conversion ratio (Eq. 1) for a given reactor, all the neutrons are accounted for and the reactor (in the assumed operating condition) is just critical. If the two values do not agree, then there is an excess or a deficiency of neutrons and the reactor is either supercritical or subcritical. Thus the reactor designer must ensure that the potential conversion ratio and the actual conversion ratio are equal and cannot arbitrarily use enrichment as a means of increasing conversion ratio unless he is able to adjust other elements of the core composition to take advantage of the neutrons potentially available.

Obviously, the last term of Eq. 2 depends only on fuel enrichment, in the manner shown by Fig. 3. The term involving p depends primarily on the ratio of U^{238} to moderator and to some extent on the lattice geometry. Figure 4 is a plot of $\eta_{25}\epsilon\,(1-p)$ as a function of the volume ratio of H_2O to UO_2, as reported from experimental results in Ref. 13. Three different lattice arrangements are included, but the effect of these variables is not large. Referring to curves 2 and 3, it is evident that at an enrichment of 3 per cent one gets a contribution of only approximately 0.13 to the conversion ratio as a result of thermal neutron absorption, and if, for example, one desires a total conversion ratio of 0.60, the contribution of the resonance absorption must be 0.47, and the H_2O to UO_2

* It is assumed here that the fuel consists of U^{235} and U^{238}. Extension to the thorium case is obvious.

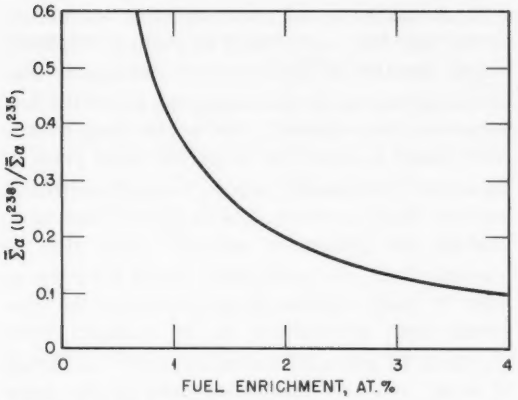

Figure 3—Effect of fuel enrichment on the thermal-neutron contribution to conversion ratio.

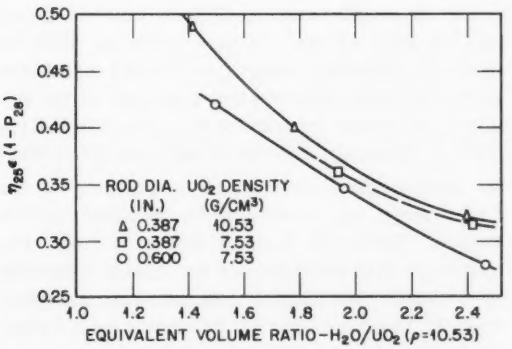

Figure 4—Effect of moderator to fuel ratio on resonance contribution to conversion ratio for cylindrical rods of UO_2 in H_2O (leakage neglected). These data were extracted from the experimental results of Ref. 11. The volume ratios have all been converted to equivalent volume ratios for UO_2 of density 10.53 g/cm³.

ratio must be about 1.5 (or somewhat less when the effect of fast leakage is included).

With the above discussion as background, it is instructive to consider the compositions, enrichments, and conversion ratios quoted for various current H_2O reactor designs, as tabulated in Table II-3. It must be recognized that in practical reactor designs the fuel-moderator lattice is not the ideal one discussed above but is modified by concentrations of the moderator in control-rod channels and between fuel subassemblies and by the concentration of structural materials in subassembly walls, etc. Furthermore, there is not universal consistency in the specification of conversion ratio, and hence the values tabulated may not be strictly comparable. Two exceptions must be noted immediately. The conversion ratio of EBWR suffers because neutron leakage is relatively important in the case of such a small reactor. The Consolidated Edison reactor, in addition to employing thorium, whose nuclear properties are different from those of U^{238} (see discussion below), employs burnable poison as a means of shim control. The burnable poison represents absorber which is parasitic in the sense that it does not contribute to the production of new fissionable material, but it does accomplish one of the major objectives of a fertile material in extending the reactivity lifetime of the fuel.

With the above exceptions, the tabulated reactors are qualitatively in agreement with the preceding discussion. Thus the Yankee reactor, by employing rather high enrichment and a quite low moderator to fuel ratio, achieves a reasonably good conversion ratio despite its use of

Table II-3 COMPOSITIONS AND CONVERSION RATIOS OF SEVERAL H_2O REACTORS

	Yankee	Consolidated Edison	USSR	EBWR	Dresden	Belgo-nucleaire
Fuel material	UO_2	$ThO_2 + UO_2$	UO_2	U metal	UO_2	UO_2
Jacket material	Steel	Steel	Zr-Nb	Zircaloy-2	Zircaloy-2	Steel
Volume H_2O / Volume UO_2 (or $ThO_2 + UO_2$)	1.44	1.54	1.90		2.17*	3.08*
Enrichment, %	3.03 (initial)	5.3 (initial)	1.5 (equilibrium feed)	1.44 (initial)	1.5 (initial)	2.5 (initial)
Conversion ratio	0.691 (initial)	0.5	~0.75† (average over fuel cycle)	0.6	0.7 (initial)	0.56 (initial)

*Measured when no steam voids present.
†Calculated from the fuel-cycle data of Ref. 12.

steel fuel-element jackets. The Russian reactor, however, employs zirconium alloy jackets and is able to attain a good conversion ratio at lower enrichment. When the boiling reactors (Dresden and Belgonucleaire) are considered, the conversion ratios attained for the two types of fuel jackets are somewhat lower, as a result of higher H_2O to UO_2 ratio. It is characteristic of present designs of boiling reactors that they do not employ extremely low moderator to fuel ratios because the ratio is chosen to yield a low (but negative) steam void coefficient of reactivity. Thus the boiling reactors are subject to an additional design condition which restricts to some degree the range of attainable conversion ratio.

The conversion ratio determines the net burn-up of fissionable material per unit of energy generated and therefore the net burn-up contribution of the fuel cost. More important than the direct monetary value of the fuel produced, however, is its value in preserving reactivity as the original fissionable isotope is burned, thus allowing long fuel lifetime. As a general principle of economic design, it can be said that the goal of the physics design of the reactor should be to provide a reactivity lifetime equal to the "metallurgical" life of the fuel elements. The latter lifetime, for a given basic fuel-element design, is determined either by radiation damage, in which case it is specified in terms of the specific energy production (Mwd/ton), or by corrosion or related effects, in which case it is determined simply by the elapsed time in the reactor. In neither case is the metallurgical life of the fuel affected directly by the fractional burn-up of contained fissionable isotope. Consequently, a fuel element of relatively high enrichment, during its metallurgical lifetime, will experience a lower fractional depletion of its original fissionable isotope than will a similar element of lower enrichment. For example, a more highly enriched element will lose reactivity less rapidly with exposure than will a less highly enriched element operating at the same conversion ratio. It is this circumstance, plus the effect of enrichment on the potential conversion ratio, which makes it feasible to design reactors for relatively long reactivity life even when rather highly absorbing structural materials, such as stainless steel, are employed.

It should be noted, however, that when relatively high fuel enrichment is used, a relatively small fraction of the converted fissionable material is burned in situ during the life of the fuel element. Consequently, the net burn-up cost in such cases is sensitive to the buy-back price of converted fissionable isotope. In a self-sufficient nuclear fuel economy (one in which there is no market for plutonium and U^{233} other than as reactor fuel), the attainment of low net burn-up cost in such reactors would depend on the economic use, somewhere in the nuclear power complex, of recycled plutonium or U^{233}, although it need not necessarily be used in the same reactor which produced it.

When the conversion ratio is too low to keep the reactivity from dropping with fuel exposure (as it is in all existing H_2O reactor designs), one or both of two possible methods must be used to maintain reactivity. Either provision must be made for frequent refueling of the reactor, or excess reactivity must be provided (by overenrichment), and the excess reactivity must be compensated during the early life of the fuel charge by control rods or other control means. Since the high-pressure vessels required by H_2O reactors are not easily adaptable to frequent opening for fuel changes, the tendency is to employ partial reloading on a rather limited scale and to build-in fairly large initial excess reactivity for fuel burn-up. The use of high excess reactivity imposes penalties of two types. The first is a change in power distribution over the time interval between reloadings as the amount of control required gradually decreases. This variation in power distribution contributes to the maximum to average power ratio and may have an important effect on the maximum useable power density. The second effect is that the control elements, at least all of those which have been incorporated into existing designs, absorb neutrons parasitically and thereby decrease the potential conversion ratio. Thus to some degree the control elements defeat their own purpose of extending fuel life by wasting neutrons and decreasing the attainable conversion ratio.

Table II-4 lists the proposed schemes of reactivity control and fuel reloading for existing water reactor designs, in so far as they can be inferred from the published information on these reactors.

Table II-4 FUELING PROGRAMS AND CONTROL METHODS FOR SEVERAL H_2O REACTORS

Reactor	Fuel life (Mwd/t fuel)	Reloading time (Full power years)	Fraction reloaded	Reactivity life (Mwd/t fuel)	Excess k for operation (% k)	Reactivity control system	Fuel unloading system
Yankee	6,550	0.57	0.5	3,275	6.7 (for non-equilibrium core)	Absorber (Ag-In-Cd) control rods. Soluble poison for hot to cold shutdown.	Remove pressure vessel top. Flood top of reactor with 25 ft of borated water. Remove individual elements to storage pit under water.
Dresden	10,000	0.5	0.2	2,000	5.7 (for non-equilibrium core)	Absorber (1.52% B-SS) rods.	Remove top of pressure vessel. Flood refueling canal with 25 ft of water. Remove individual elements under water.
Consolidated Edison	18,000	1.65	1.0	18,000	7.0 (?)	Absorber (Hf) control rods. B^{10} burnable poison in core. B^{10} soluble poison in coolant.	Remove top of pressure vessel. Flood refueling canal with water. Remove individual elements under water.
Belgonucleaire	10,000	0.5	0.2	2,000	1.25 (for equilibrium core)	Absorber (2% B-SS) control rods. Fixed absorber shims.	Remove pressure vessel top with flooded reactor vessel. Shielded coffin unloader removes fuel through plug in reactor tank.
Russian	8,300	0.4	0.33	2,760	(?)	Combined fuel removal and absorber insertion.	Remove pressure vessel top and shield plug. Flood top of reactor to give 16 ft of water shielding. Remove individual elements with crane.

Thorium as a Fertile Material

Although the fuel cycles involving thorium as fertile material have received theoretical attention in the past, it has remained for the experimental and theoretical studies connected with the design of the Consolidated Edison reactor to explore the performance of such cycles within the framework of the present technology of solid-fuel–water reactors. A report of these studies is contained in Ref. 14. The reference contains useful experimental data and, in addition, makes these points concerning the use of thorium as fertile material in H_2O-cooled and -moderated reactors:

1. Because thorium has a considerably higher thermal absorption cross section than U^{235} (7.0 barns vs. 2.75 barns), the thermal contribution to the conversion ratio, at a given ratio of fertile to fissionable material, will be proportionately greater (Fig. 3). Thus if a high density of fissionable isotope is desired in the reactor in order to increase the potential conversion ratio (Eq. 1), the amount of thorium which must be packed into the reactor core to absorb the excess neutrons is less than the amount of U^{238} which would be required for the same purpose. In other words, in a reactor having a given concentration of parasitic absorber, a given value of conversion ratio can be achieved in the thorium reactor at a lower ratio of fuel to moderator than would be the case in the uranium reactor.

2. Because higher concentrations of fissionable isotope can be used in the reactor core without sacrificing actual conversion ratio, the fractional depletion of fissionable isotope for a given fuel exposure (Mwd/ton) is reduced. As previously mentioned, this effect reduces the rate at which reactivity is lost with fuel exposure (at a given conversion ratio) at the expense of a larger inventory of fissionable isotope and a smaller consumption of converted fissionable isotope in situ.

Unfortunately the application of the above arguments to the Consolidated Edison reactor is obscured by the use of burnable poison in the reactor. They are, however, practical considerations which must be added to those that have long been recognized (higher η of U^{233}; lower fast fission effect in thorium; relatively long half life of the protoactinium precursor of U^{233}) as affecting the usefulness of thorium as a fertile material.

Boiling and Nonboiling Reactors; Effects on Thermal Performance

Any discussion of the thermal performance of water-cooled reactors must take account of whether or not the reactors are designed for boiling of the coolant. It has often been pointed out that the current designs of boiling and nonboiling pressurized water reactors represent merely the two extremes of a range of water-cooled reactors which may employ boiling to a greater or lesser degree. This point of view has been explored in the USSR in a study which covers nonboiling reactors, direct-cycle boiling reactors, and a number of intermediate types.[15]

It is quite possible that one of the intermediate types will eventually prove most attractive, but the present discussion can best be directed toward the two extreme cases, not only because they emphasize the peculiar characteristics of the types but also because current designs fall near the two extremes.

Many of the differences between boiling and nonboiling reactors cannot be evaluated in general terms but only in so far as they contribute to the total performance of a particular reactor design. Among the more important differences are the following. Quite obviously boiling reactors can generate higher pressure steam for a given reactor pressure. Boiling reactors are generally given credit for greater simplicity and for greater safety against accidental reactivity additions. On the debit side they impose more restrictions on the physics design, as discussed briefly above, and are perhaps limited to lower power densities. Two questions which have not yet been completely answered for the boiling reactor are the seriousness of the increased radiolytic decomposition of water which occurs in such reactors and, for the direct-cycle reactors, the magnitude of the problem of radioactive contamination. Those of the above considerations which are amenable to some quantitative treatment will be discussed briefly in the following paragraphs.

Steam Pressures

The rather large differences between the steam pressures attainable at the turbine with boiling and nonboiling reactors is due to the characteristically steep rise of water vapor pressure with temperature in the range of interest. If the reactor is to be so designed that no boiling occurs at the surface of the hottest fuel element, then the reactor must be pressurized to a level equal to or greater than the vapor pressure of water at the surface temperature attained by the hottest fuel element under the most adverse operating conditions. This temperature will be higher than the secondary steam temperature by some amount, ΔT, which will depend on the film temperature drop at the fuel element and boiler tube surfaces, on whatever hot channel factors are appropriate in the reactor, and on any additional allowances which may be made for overshoots of reactor power. If a value of 150°F is taken for ΔT, a relation between secondary steam pressure and reactor pressure results which comes close to fitting current nonboiling designs. This relation is plotted in curve A of Fig. 5.

In the indirect-cycle boiling reactor, the only contribution to ΔT arises from the heat-transfer temperature drop in the secondary boiler. Only one example of such a design is available in detail (Belgonucleaire).[16] A value of 48°F is appropriate for that design, and a curve based on a constant value of 48°F has been drawn as curve B of Fig. 4. In the direct-cycle case the steam pressure is very nearly equal to the

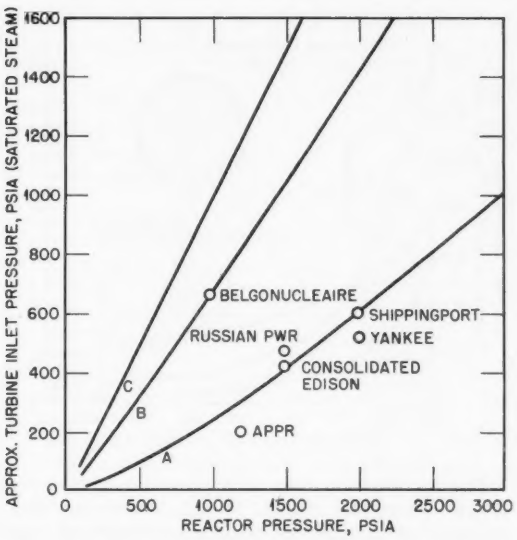

Figure 5—Possible relations between turbine inlet pressure and reactor pressure for boiling and nonboiling reactors. Curve A, nonboiling reactors (ΔT = 150°F). Curve B, indirect-cycle boiling reactors (ΔT = 48°F). Curve C, direct-cycle boiling reactors (Δp = 25 psi).

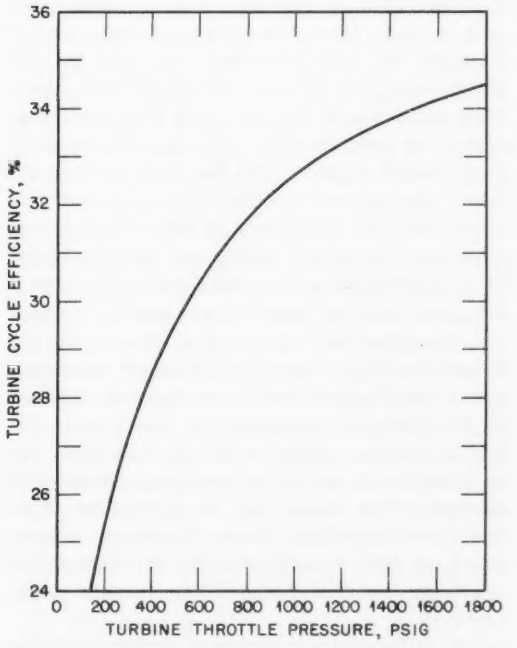

Figure 6—Calculated turbine cycle efficiency as a function of pressure for saturated steam. Five-stage regenerative cycle; optimum feedwater rise; 1.5-in. Hg back pressure; 100- to 200-Mw capability. (Reproduced from Ref. 17.)

reactor pressure, being lower only by the pressure drop in the moisture separator and the piping. A constant value of 25 psi has been assumed in curve C of Fig. 5.

The question of how much the higher steam pressure attainable with the boiling reactor is worth, or alternatively how much advantage results from the lower reactor pressure of boiling reactors, is a difficult one. For large central station applications it may be predicted that boiling reactors will operate in the relative high pressure range, 1000 psi and above, because of the heat removal advantages which result from high pressure. Thus a typical comparison might be between a boiling reactor operating at 1000 psi and nonboiling reactor operating at about 1500 psi and producing steam at a pressure of about 500 psi. For whatever the comparison is worth, a curve of efficiency as a function of pressure for saturated steam, as reported in Ref. 17, is given in Fig. 6. As steam pressure is raised the moisture problem becomes more important, and it may be expected that superheating for boiling reactors, either by chemical or nuclear means, will continue to be attractive in order to realize the full potentialities of the reactor type.

Power Density and Specific Power

The power density attainable in the reactor is important because it determines the total output from a plant of given size and therefore the specific capital cost. It is also important because there are practical limitations on pressure vessel size, which, along with the power density, set the limit on the maximum output of a single reactor plant. The specific power (megawatts per ton of fuel) is important because it determines the inventory of fabricated fuel necessary for a given power output and thereby determines the contributions of the use charge and the capital charge to the total fuel cost. As a class, water reactors have relatively high power densities and specific powers. However, improvements in these characteristics are worth while because the high operating pressures of water reactors place important limits on pressure vessel size and because the construction of water reactors, when considered on a volumetric basis, is expensive.

Early in the development of the boiling reactor, it was anticipated that power density might be limited by reactor instability. The

performance of the EBWR, BORAX-IV, and the Vallecitos Boiling Water Reactor have demonstrated, however, that the early hopes for improved stability at higher reactor pressures were fully justified and, furthermore, that the use of fuel elements of long thermal time constant (typical of those used in central station power reactors) also makes a substantial contribution to stability. Investigations of reactor stability have been summarized in Refs. 18 and 19, and more recent data are contained in Refs. 20 and 21. Supporting information from laboratory tests is contained in Ref. 22. It now appears that there is a rather broad area of boiling reactor design in which the power of the reactor will not be directly limited by considerations of instability. Indirectly, the demands of stable design, which dictate the desirable range of steam void coefficient of reactivity, will affect the useable range of fuel to coolant ratios and thus may affect the maximum attainable power density.

Aside from stability considerations, the limitations on power density, which apply to both boiling and nonboiling reactors, are the limitations on heat transfer to water (which if exceeded cause burn-out of fuel elements) and the limitations on heat conduction in the fuel element (which if exceeded cause excessive temperatures at the center of the element). Although these limitations are hard to discuss in terms of an arbitrary fuel element, a consideration of their relations to the fuel elements currently in use can clarify the basic relations.

At given coolant conditions (flow velocity, subcooling, etc.), the occurrence of fuel-element burn-out is determined by the heat flux at the fuel jacket–water interface. At a given surface heat flux the central temperature of a fuel element of given shape and composition depends on the size of the element, and for a given maximum permissible temperature the limiting surface heat flux can always be increased by decreasing the element size. Curve A of Fig. 7 shows, for a typical metal-jacketed UO_2 fuel element, the calculated surface heat fluxes which will result in a central temperature of 4800°F, as a function of fuel-element diameter. It is generally accepted that heat fluxes as high as 400,000 Btu/(sq ft)(hr) can be used in boiling reactors without danger of burn-out (the calculated maximum heat flux in EBWR

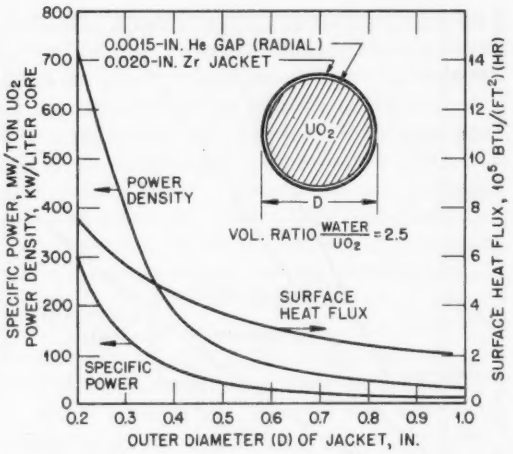

Figure 7—Surface heat flux, specific power, and power density as limited by UO_2 centerline temperature of 4800°F. All values are maximum (not average) values. T (ambient) = 550°F; h (water film) = 5000 Btu/(hr)(sq ft)(°F); k (UO_2 = 1.0 Btu/(ft)(hr)(°F); k (Zr) = 8.2 Btu/(ft)(hr)(°F); k (He) = 0.14 Btu/(ft)(hr)(°F).

when operating at 60 Mw (thermal) is 464,000 Btu/(sq ft)(hr). From Fig. 7 it is evident that such a heat flux will require a fuel-element diameter at least as small as 0.44 in. If a higher heat flux of, say, 600,000 Btu/(sq ft)(hr) were regarded as permissible in a particular nonboiling reactor design, the fuel-temperature limit would require that the fuel-element diameter be reduced to about 0.28 in. It should be noted that the reduction of fuel-element size increases the power density and specific power not only because higher heat fluxes may be used but also because more heat-transfer surface can be packed into a given core volume. Curves B and C of Fig. 7 show the attainable maximum power densities and specific powers as functions of fuel-element diameter, on the assumption that a constant ratio of H_2O to fuel (UO_2) can be maintained as the fuel-element diameter is changed. This assumption is limited by practical considerations since clearances become small as fuel-element size is decreased, and tolerance specifications must become more rigorous.

The differences in thermal performance between boiling and nonboiling reactors, as they evolve in practice, can be examined by means of Table II-5. In this table the Shippingport reactor and EBWR are included simply for

Table II-5 THERMAL OUTPUT CHARACTERISTICS OF H_2O REACTORS

	Shippingport*	Yankee	Consolidated Edison	USSR	EBWR	Dresden	Belgonucleaire
Thermal output, Mw	120†	392	500	760	60‡	626	242
Fuel load, tons of oxide	16	25.1	22.2	44		65.8	23.53
Oxide diameter, in.	0.360	0.290	0.312	0.346	Metal fuel	0.494	0.384
Jacket thickness, in.	0.026	0.021	0.020	0.028		0.030	0.015
Maximum heat flux, Btu/(sq ft)(hr)	~354,000	450,000	450,000	443,000	464,000	277,000	365,000
Average heat flux, Btu/(sq ft)(hr)	~48,300	87,000	112,500	83,000	136,500	94,700	79,300
Maximum heat flux / Average heat flux	7.33	4.18	4.00	5.34	3.40	2.90	4.60
Surface/volume ratio of fuel	11.1	13.8	12.8	11.6	~8.5	8.1	10.4
Specific power, Mw/ton of oxide	7.5	15.62	22.5	17.3		9.5	10.3
Volume H_2O / Volume oxide	1.45	1.44	~1.54	1.90	Metal fuel	2.17	3.08
Average power density, Kw/L of core	23.3	60	66	43	52	28.9	25.5

*Quantities apply to blanket only.
†225 Mw total, 105 in seed, 120 in blanket.
‡Experimental operation.

reference. The comparison which may be considered as typical of the present state of the art is between the Yankee, Consolidated Edison, and Russian reactors as representatives of nonboiling designs, and the Dresden and the Belgonucleaire reactors as representatives of boiling designs. The maximum heat fluxes in the nonboiling reactors are higher by from 30 to 60 per cent than those used in the boiling designs. The average heat fluxes, however, in the two cases are not greatly different. It is difficult to say whether there is an inherent difference in the degree of power flattening attainable in the two reactor types; certainly it cannot be concluded that the differences in maximum to average heat flux ratios shown in the table have general meaning, because the reactor designs have been made at different times and by different groups and have been carried to different degrees of refinement. Whatever the general significance of the maximum to average ratio, it is clear that the higher specific powers exhibited by the nonboiling reactors in the table are due largely to the greater subdivision of the fuel (higher surface to volume ratio). The higher specific power, when combined with the higher ratio of fuel volume to water volume in the nonboiling reactors, leads to power densities which are roughly twice as high as those exhibited by the boiling reactors. Whether this degree of difference in power density will continue to be characteristic, or to what extent it will be reduced or eliminated as later boiling designs reflect the more recent experimental results, is not presently evident. It must, of course, be recognized that the power density which characterizes a given design results from an economic optimization process and that the economic incentives toward high power density are not as great in boiling as in nonboiling reactors.

Water Dissociation

One of the early observations on experimental boiling reactors was that the net radiolytic decomposition of the water, which in high-temperature nonboiling reactors is very small, in boiling reactors is a significant effect. The latest quantitative data from EBWR[20] and the Vallecitos[21] reactor are given in Fig. 8. It is evident that the net decomposition rate increases approximately linearly with power and is decreased by an increase in reactor pressure. The decomposition rates at 600 and 300 psig in EBWR correspond respectively to G_H, values of 0.15 and 0.21 molecules of hydrogen produced per 100 ev of radiation energy absorbed in the reactor water. These G values were calculated on the assumption that 8 Mev of radiation energy is absorbed by the water per fission.

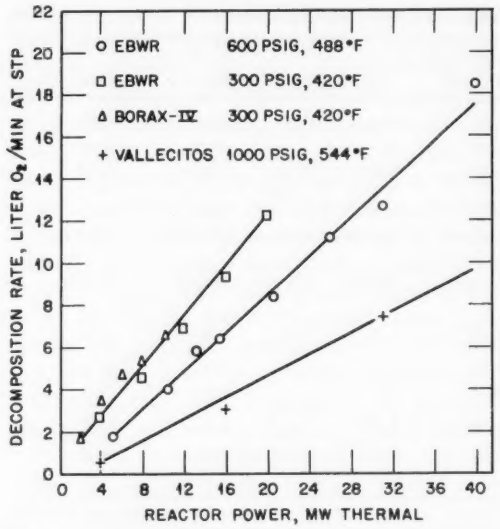

Figure 8—Water decomposition rates in experimental boiling reactors. EBWR and BORAX data from Ref. 20; Vallecitos data from Ref. 21.

In the direct-cycle boiling reactor, the dissociation gases come off in the normal condenser air ejectors, and the only concern is whether the presence of the gases in the turbine and condenser produces any deleterious effect on the materials of this equipment. No effects have yet been observed in EBWR which could be attributed to the presence of the dissociated hydrogen and oxygen. In the case of the indirect-cycle reactor, the gases are not as easily disposed of. In the Belgonucleaire design, radiolytic gas removal is accomplished by condensing only 90 per cent of the primary steam in the main primary–secondary heat exchanger. The uncondensed steam, which contains all the noncondensable gases, is bled off and is passed through further small heat exchangers where its energy is transferred to the secondary loop (partly in economizer sections), and the temperature is reduced to a level at which the noncondensable gases can be bled off through ejectors without the loss of an important fraction of steam.

Radioactive Contamination

Operation of the EBWR has now continued long enough to give some confidence that the contamination problems in normal operation will not be severe. During operation the principal activity is from N^{16} resulting from the $O^{16}(n,p)N^{16}$ reaction. This activity, half life of 7.4 sec, disappears rapidly after shutdown. An inspection of the EBWR turbine, after 2840 hr of operation covering a period of one year, revealed the activities tabulated in Table II-6. The principal source of radioactivity was found to be Co^{58}. Other isotopes which were identified in small amounts were Fe^{59}, Cr^{51}, Ba^{140}, and La^{140}.

Full-flow filters are used on the EBWR condensate stream. The maximum activity at 2 in. from the unshielded surfaces of these vessels was 15 mr/hr during operation at 20 Mw (thermal). An analysis of the insolubles on 1 sq in. of a 2 μ filter, after 230 hr of operation, indicated 67 per cent iron, 30 per cent nickel, and a 3 per cent mixture constituting manganese, chromium, and molybdenum.

The EBWR water is purified by continuously acting ion-exchange columns on a side stream, operating at reactor pressure. During normal operation the activity of the resin column is 10 r/hr through the $\frac{1}{2}$-in.-thick steel vessel wall. The main radionuclides present on the ion-exchange resin and associated filters are Co^{58}, Ba^{140}, La^{140}, Fe^{59}, and Co^{60}. Cr^{51} was found only in the main resin, presumably in the form of chromate ions.

Table II-6 EBWR TURBINE ACTIVITIES, 10 DAYS AFTER SHUTDOWN[20]

Location	Activity at 2 in., mr/hr
Steam nozzle inlets	5
Admission valve stems	3
No. 1 HP steam seal	2
Throttle valve inlet	2
First row of blades	0.2
Remaining rows of blades	0.2 (max.)

A more important question is that of the contamination which may be expected when fuel-element failures occur. In March 1958, the BORAX-IV reactor was operated with a fuel defect of unknown origin, apparently involving a number of individual fuel elements. These fuel elements consisted of UO_2-ThO_2 pellets in aluminum jackets with a lead bond between pellets and jacket. During this period the reactor was operated at powers up to 20.5 Mw (thermal). The activities observed at various points in the reactor system are tabulated in

Table II-7 BORAX-IV SURFACE ACTIVITIES IN MR/HR DURING OPERATION WITH FUEL DEFECT[20]

	Before operation	2.4 Mw (thermal)	6 Mw (thermal)	17 hr after shutdown	Normal activities at 12 Mw (thermal)
Steam pipe turbine bldg.	0	250	400	0	200
Turbine exhaust casing (external)	0	150	190	0	50
Turbine exhaust casing (internal)	0.7 γ 15 $\beta + \gamma$			0.7 γ 15 $\beta + \gamma$	
Condenser hotwell	0	2,200	5,000	1.5	500
Air ejector after-cooler	0	15,000		0	420
Air ejector discharge (2- by 2- by 1-ft AEC type filter)	0	30,000	50,000	1	

Table II-7. At 2.4 Mw (thermal) the fission gas release from the air ejector contained 3.0 ± 1.5 curies/min of Xe^{138} and 0.6 ± 0.1 curies/min of Kr^{88}. The total gas flow from the air ejector was 183 liters/min.

It is stated that during the operation with the defective fuel elements the gross activity in the reactor water showed no substantial increase over normal operating conditions. This observation would indicate that only gaseous fission products were released by the oxide fuel elements to an important extent; it is highly encouraging as an indication that high levels of post-shutdown contamination are not to be expected.

Recent Innovations and Future Trends

The designs of the United States H_2O reactors presently under construction have been reasonably well known for some time, and consequently it is only from the recently published Russian work that novel design features can be reported. Although the over-all picture of the Russian PWR design seems rather similar to those which are current in the United States, there are a number of features which differ significantly from American practice.

The pressure vessel of the Russian PWR is constructed of high-strength low-alloy steel lined with a corrosion-resistant stainless-steel clad, which may be contrasted with the carbon steel usually used in reactor vessels in the United States. The Russian vessel uses a wall thickness of only 3.94 in. in the straight sections of the vessel, which has an inside diameter of 11.7 ft and a working pressure of 1500 psi. It is reported that the Consolidated Edison reactor, also for 1500 psi working pressure, but of 9.75 ft I.D., has a wall thickness of $6^{15}/_{16}$ in. (SA212, Grade B).[8] Obviously the question of pressure vessel material and allowable design stresses is an important one for pressurized water reactors, inasmuch as these considerations set the limitations on reactor size and operating pressure.

The control rods employed in the Russian PWR are of the MTR type, having a fueled section and an absorber section, so that absorber is replaced by fuel as it is withdrawn from the reactor. This type has also been used in the APPR in the United States. Although this type rod poses some design problems connected with the cooling of the fueled portion, its use probably results in performance advantages. Individual rods of this type can be made to control larger amounts of reactivity. If the rod is made in the same size and shape as the regular fuel element in the core, the rod does not interrupt the regular pattern of the fuel subassemblies, and consequently it probably allows more efficient use of the core volume for heat transfer. The rod type should also reduce the width of the water channels associated with clearance between the rod and fuel elements and therefore reduce the magnitude of local flux peaking.

As mentioned previously, the Russian fuel elements are jacketed in a zirconium-niobium alloy rather than Zircaloy-2 as is the practice in the United States. The advantages claimed for the alloy have already been mentioned. A departure in fuel-element construction which is perhaps more significant is the Russian use of fuel elements of full-core length, which are

spaced in the fuel subassembly by zirconium spacer grids which make only frictional contact with the fuel-element jackets. This construction has been avoided in the United States because of the possibility of fretting corrosion at the contact between the spacer and the fuel element. The statement that the Russian elements have been tested and show no fretting corrosion is of considerable interest. The rather complex structures which have been used in some U.S. reactors for the fuel subassemblies are expensive and lead to rather poor hydraulic performance.

With regard to design trends, one of the more important questions is that of zirconium vs. stainless steel. It is curious that the United States, which first developed zirconium and the Zircaloys, has lately favored stainless steel for fuel-element jackets, whereas the USSR, which began late in the development of zirconium, intends to use the material for its pressurized water reactor and, in using it, will apparently achieve a nuclear performance which is quite considerably better than that exhibited by any of the U.S. reactors. Three factors have been involved in the United States trend away from zirconium: the fear of zirconium-water reactions in the event of melting of fuel-element jackets; the high cost of manufacturing the early zirconium fuel elements; and the establishment of prices for fuel enrichment and buy-back prices for plutonium, which make the net fuel cost rather insensitive to enrichment in the enrichment range above 1.5 or 2 per cent. More recently the price differential between zirconium-jacketed and steel-jacketed fuel elements appears to be less than was originally supposed. If it is assumed that this trend will continue as experience in fuel-element fabrication builds up and that eventually plutonium prices will seek a level determined by the value of plutonium as fuel, then it may be expected that the economic consideration will tend to reverse the trend toward the use of steel. The question of metal-water reactions is one which needs further investigation.

On the question of boiling vs. nonboiling reactors, it may be said that almost all recent experimental data have been encouraging with respect to the capabilities of boiling reactors and that more design activity has been evident recently on new boiling reactors than on new nonboiling reactors. As has been mentioned before, it is probable that the two types will tend to become more similar, at least to the extent of toleration of local boiling in the PWR types. Although good progress has been made toward an understanding of the heat transfer, steam flow, and stability questions in boiling reactors, boiling reactor design is still handicapped by uncertainties regarding the design limits for this reactor type. Data are particularly needed on reactors with forced circulation for, whereas it is possible to set rather reasonable lower limits for the permissible design quantities in such reactors, the degree to which these estimated lower limits may safely be exceeded is not well known. As a general trend in boiling reactor designs, it may be predicted that design pressures for plants in the central station power range will tend to be high (e.g., 1000 psi or above), and it may be expected that there will be an increased interest in nuclear superheat for reactors of this type.

In the field of capital costs, rather large improvements might result from higher power densities if the higher densities could be achieved without such expensive expedients as very fine subdivision of the fuel. The quite high maximum to average power ratios in water reactors are striking, and it is to be hoped that some improvement can be made in this characteristic. To some extent this characteristic is influenced by fuel lifetime or, more precisely, reactivity lifetime. Long exposure of the fuel between reloadings requires high excess reactivity and strong control elements. The changes in position of control elements and the uneven burnup of fuel as irradiation proceeds cause changes in the flux distribution with time which greatly complicate the problem of maintaining a flat power distribution. There would seem to be a large incentive toward the development of economic means for partial reloading of fuel, and perhaps fuel reshuffling, at rather short intervals.

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The Calder Hall type gas-cooled reactor has been developed by the British to the point where it is said to be an economically practical type for base load operation in the United Kingdom. The French nuclear power effort has also centered around the same basic reactor type. Unquestionably, much of the attraction of this reactor type lies in the fact that it is the only type using relatively familiar materials which can be constructed and operated without isotope separation of some kind, either for the fuel or for the moderator. The requirement of operation on natural uranium is a very restrictive one. With graphite as moderator, it requires that uranium be used in a very dense form and that parasitic absorbers be held to a very low level. The only form of uranium which will provide sufficient reactivity in the graphite-moderated reactor appears to be uranium metal.

Despite the restrictions imposed by natural uranium, a combination of materials has been evolved which has made possible the British and French reactors currently under construction and in operation. The basic materials are graphite as moderator, uranium metal as fuel, magnesium alloys as fuel jackets, and carbon dioxide as coolant. Although considerable variety has been evident in the solutions of the engineering problems of such reactors, the restrictions imposed by the materials set rather rigorous limitations on performance. Rapid progress has been made through careful and ingenious design in approaching the boundaries set by these limitations and has emphasized the need for circumventing the limitations by basic changes in reactor materials.

Performance Characteristics

The important performance characteristics of the natural-uranium graphite-moderated reactor can be summarized rather easily. The physics of the reactor design is dominated by the need for making the reactor critical with the natural fuel. The fuel to moderator ratio must be chosen near the value which will give maximum reactivity for the reactor geometry in question. Since it will usually be desirable to flatten the power distribution in the reactor, which is accomplished at the expense of increased neutron leakage, the maximum-reactivity lattice will not be far from that giving maximum material buckling. If shifts away from the maximum reactivity lattice are made, they would normally be made in the direction of smaller lattice spacing in order to make possible higher power density. The degree to which such variations are possible is rather small, as is evident from the nearly standard values of lattice geometry in all the British gas-cooled reactors. Since there is little variation in lattice characteristics, there is little variation also in conversion ratio. The conversion ratio of the graphite-moderated natural-uranium reactor is necessarily relatively good, lying in a range between 0.80 and 0.85. In this range of conversion ratios, and with the amounts of excess reactivity which can be attained in such reactors, reactivity lifetimes can be attained which are probably longer than the metallurgical lifetimes of the fuel elements. The latter do not appear to be longer than 3000 Mwd/ton with present technology.

The most obvious limitation on the thermal performance of the reactors is the temperature limit of the magnesium alloy jacket. This limit is about 850°F. Working to this limit, coolant exit temperatures in the range 700 to about 750°F can be attained, the higher values being attained at the expense of some decrease in specific power. The specific power of the reactors is relatively low, ranging from 1.52 Mw/ton of fuel for the Calder Hall reactors to 2.29 Mw/ton of fuel for Hinkley Point. In Calder Hall the limitation on specific power arose from gas heat transfer and heat transport limitations. It has been stated[1] that, with the construction of thicker-walled pressure vessels and improvement in heat transfer design, the limitation is no longer due to gas heat transfer

considerations but to excessive central temperatures in the fuel element. When the central temperature becomes too high, the metallic fuel element is subject to swelling as gaseous fission products accumulate under irradiation. Central temperatures could be reduced at a given specific power by decreasing the size of the fuel elements. The same reference[1] states that it does not seem that any worthwhile thinning or subdivision of the fuel structure is possible while using natural uranium in these reactors (because of the loss of reactivity which accompanies the subdivision of the fuel, due to increased resonance neutron absorption in U^{238}). Taken in its intended context, this statement evidently means that no major improvements can be made. It may well be that smaller but significant increases will be made through relatively small decreases in fuel-element size.

The limitations of the natural-uranium graphite reactor, which appear to be inherent, are then: relatively low temperatures, low specific power, and rather short fuel lifetime. Agreement seems to be general that the most attractive way of improving these characteristics is through the use of uranium oxide as fuel. Adoption of this fuel implies that enriched uranium will be used. Once the decision has been made to adopt this course, the procedures for improving reactor performance are evident. It is first necessary that fuel jackets be developed which will withstand considerably higher operating temperatures. The oxide fuel can be operated at any temperature up to the limiting temperature of the jacket. Subdivision of the fuel will of course be necessary but is feasible if enriched uranium is used. Specific power can be increased also by subdivision and if necessary by decreasing over-all reactor size to make possible higher gas pressure with given pressure-vessel wall thickness. At some point, specific power per se ceases to be of great importance, and power density, which affects the capital cost of the plant, becomes the major consideration. Further enrichment can make possible increases in power density through decreases in moderator to fuel ratio.

Once the oxide fuel–enriched fuel approach has been adopted, the remaining important question is what to use for fuel jacket. Among the low cross-section materials there appear to be only two possibilities: beryllium and zirconium; and of these there are serious reservations as to the ability of zirconium to operate in very hot gases. Beryllium appears very promising from the standpoint of performance, but the fabrication of beryllium-jacketed fuel elements presents problems. Both the British and French[2] appear to have confidence in the development of such elements, at least of the unfinned variety.

The alternative to the use of low cross-section fuel jackets is the use of jackets made of medium cross-section materials, and of such materials stainless steel is an obvious one. If it is assumed that beryllium jackets will be developed, then the choice between beryllium and steel involves balancing the quite considerably better neutron economy achievable with the beryllium jackets against a somewhat higher temperature capability of the steel, a presumably lower fabrication cost of the steel, and possibly differences in performance of the two materials which may show up in actual use. In a nuclear fuel economy which is geared to the large-scale separation of isotopes, it could be that the advantages of steel would make it attractive economically at least for a number of years. The considerations affecting neutron performance with such fuel would be much like those discussed in the preceding section for steel-jacketed fuel elements in water-moderated reactors. As was pointed out there, it is possible to achieve relatively good neutron economy with such fuels provided the enrichment is made rather high.

For a country which has accepted the necessity for fuel enrichment, but which has not already made a large investment in isotope-separation plants, the beryllium route is almost sure to appear the more attractive. The possibility of recycling plutonium to attain the required enrichment is in such a case a very attractive one, notwithstanding the known difficulties involved in plutonium recycle. If it is recognized that the recycle of plutonium will ultimately be necessary from the point of view of fuel conservation, then it may well appear that an investment in research and development and plant construction, directed toward economic plutonium recycle, may be a sounder one, even at the present time, than a large investment in the expansion of isotope-separation capability. If a fuel economy based

on plutonium recycle is contemplated, then the achievement of high neutron economy is of the utmost importance, and there is little question that the beryllium-jacketed fuel element would be the logical one to develop if at all possible. Obviously, considerations of this kind are playing an important part in the British and French developments in gas-cooled reactors. Indeed, it has been stated[3] that the high cost of enriching fuel in the United Kingdom has been a major factor in excluding H_2O reactors from the British program and in the decision to discontinue work on the sodium-graphite reactor; it is improbable that a gas-cooled effort involving a reactor of poor neutron economy would be supported in that country. The American approach, however, has been to consider steel-jacketed elements, at least for the near future.

Reactor Designs

In Table III-1 are compared prototype reactors of the graphite-moderated gas-cooled type which have been proposed in the United States and in the United Kingdom. Except for the difference in fuel jackets (and therefore in neutron economy), the characteristics of the two designs are remarkably similar. The thermal design of the British reactor appears to be somewhat more refined than that of the American one because the superior high-temperature capability of the steel jackets in the latter design does not appear to improve the performance as much as might be expected. The American design summarized is the prototype design from the recent Kaiser-ACF studies made for the AEC.[4,5] The British design is that for the Advanced Gas-cooled Reactor (AGR)[1] which is being designed and constructed as a prototype of the reactor visualized as the next major step beyond the Calder Hall type. The design criteria for the AGR are experimental utility rather than economic power generation. Thus an unflattened core design is used to give a wide range of heat generation rates in the fuel elements. Each fuel channel is connected directly to individual access ports in the pressure vessel head and is provided with complete instrumentation and gas flow control. Refueling can be carried out under full-power operation. The graphite moderator is completely replaceable through provision of a shield inside the vessel above the core. This shield permits full access to the upper head of the vessel for its removal and subsequent replacement.

The general layout of the reactor is such that the fuel channels are vertical and the gas flows upward through the core. The core, complete with its restraint gear and internal thermal shield, is contained within a vessel 21 ft in diameter and about 60 ft in over-all height. Refueling and control are from the top face. Coolant nozzles are provided in the bottom head of the vessel, giving straight-through access and enabling test loops to be incorporated. The reactor vessel is of double-shell construction. Carbon dioxide at the inlet temperature of 480°F (maximum inlet 620°F) sweeps the gap between inner and outer shells, keeping the main wall of the pressure vessel at the lower gas temperature.

The external circuit is made up of four parallel loops, each consisting of a heat exchanger and main circulator. The heat exchangers are raised in relation to the reactor itself in order to promote natural circulation of the coolant in the event of main circulator failure. Since the plant is purely experimental, a single-pressure steam cycle is used with the relatively modest steam conditions of 600 psi, 850°F at the turbine stop valve. A higher efficiency could be achieved with a more complex steam cycle of the type which would be used in the civil power stations eventually designed on the basis of AGR technology. A single turbine is used, coupled with dump condenser facilities capable of rejecting the full heat output of the plant to forced-draught cooling towers. The main circulators are electrically driven through fluid couplings giving a 4 to 1 speed range.

The safety problems of the reactor are similar to those of existing reactors in that the fuel elements are designed to retain the fission products; the reactor operates with a clean coolant and is equipped to detect and locate faulty fuel elements. In view of the experimental nature of the reactor and the fact that it is anticipated that many experiments will be carried out in it, it has been decided to design for full containment. The reactor unit is therefore enclosed in a steel container building which will accept the total contents of the CO_2-gas circuit.

Table III-1 SUMMARY OF DESIGN CHARACTERISTICS FOR THE U. K. ADVANCED GAS-COOLED REACTOR (AGR)[1] AND THE KAISER-ACF PROTOTYPE DESIGN (GCPR)[4]

	AGR	GCPR	Revised GCPR*
Reactor thermal power, Mw	100	125	84
Net electrical power, Mw	~28	44	26
Core length (excluding reflector), ft	14	14.5	17
Core diameter (excluding reflector), ft	15	12	11
Number of fuel channels	250	344	248
Reactor coolant	CO_2	CO_2	He
Gas pressure at inlet to reactor, psig	270	385	315
Gas temperature at inlet to reactor, °F	480–620 (peak)	463	510
Bulk gas outlet temperature, °F	930–1050 (peak)	1000	1050
Normal maximum fuel-element can surface temperature, °F	1110	1300	1300
Fuel material	UO_2	UO_2	UO_2
Fuel loading, metric tons	14.6	18	13
Specific power, mean, Mw/metric ton U	7.75	6.9	7.6
Specific power, peak, Mw/metric ton U	18.00		12.6
Initial U^{235} enrichment, %	~1.2	3	2
Fuel canning material	Beryllium	S.S.	S.S.
Number of control rods	25	24	29
Total reactivity controlled, %	~14		
Pressure vessel inside diameter, ft	21	18	20
Pressure vessel over-all height, ft	~60	40	46
Weight of graphite, tons	~200	170	170
Number of external heat-transfer circuits	4	2	2
Number of main CO_2 circulators	4	2	2
Refueling scheme	During operation	During operation	During operation
Steam cycle	Single pressure	Dual pressure reheat	Duel pressure reheat
Superheater outlet pressure, psig	620–670	2400	1000
Superheater outlet temperature, °F	860	950	1250
Steam pressure at turbine, psig	600	2400, 750	1250, 300
Steam temperature at turbine, °F	850	950	1000

*Latest GCPR plans call for helium as coolant.

A question which has not been resolved in the high-temperature gas-cooled program is whether carbon dioxide can be used with graphite at the high temperatures visualized. It seems clear that the chemical compatibility of the two materials is not sufficiently good for these temperatures unless a means can be found of protecting the graphite or of inhibiting the chemical reaction. There is optimism that such remedies can be found,[1] but if these do not materialize and it becomes necessary to employ helium as the coolant, a large developmental effort may be necessary to ensure that leakage of the rather expensive coolant can be kept within economic bounds, and the cost of the special design features necessary for helium containment may constitute a significant addition to the capital cost of such reactors.

An alternate approach to the development of higher-temperature gas-cooled reactors is through the use of heavy water rather than graphite as the moderator. In this case the superior nuclear properties of the heavy water make possible a reactor operating on natural uranium oxide, provided a low absorption fuel jacket is available. A version of this reactor type is under development in the United States[6] (50-Mw electrical prototype is planned), and studies are under way in the United Kingdom.[3] In both cases the reactor is of the pressure-tube type. The D_2O moderator is contained in an unpressurized tank through which pass zirconium tubes of sufficient strength to contain the high-pressure coolant gas (CO_2). The fuel elements are installed in the pressure tubes.

A D_2O-moderated gas-cooled reactor is also planned in Czechoslovakia,[7] with assistance from the USSR.[8] The plant, which is to be designed for 150 Mw (electrical), is quite different from the U. S. and U. K. concepts. It employs uranium metal as fuel but with a high degree of subdivision which lowers the temperature drop

between the center and the surface of the fuel elements. For jackets, an alloy of magnesium–1.5 per cent beryllium is said to permit operation at surface temperatures up to 500°C (932°F) in CO_2. This combination of small-diameter uranium metal rods and magnesium-beryllium jackets is expected to make possible a coolant outlet temperature of 425°C (798°F) and maximum steam temperature of 400°C (752°F). Despite the high degree of subdivision of the fuel, operation on uranium of natural enrichment is possible because of the superior moderating properties of D_2O.

The design differs further in that it employs a pressure vessel instead of pressure tubes. The vessel is to be welded up in the field from shop-fabricated ring sections. The D_2O moderator is contained in an aluminum calandria inside the pressure vessel; it is at coolant pressure (60 atm) but at low temperature. The inlet gas temperature is very low (100°C), and a small fraction of this gas is sent through the annulus between each calandria tube and its contained fuel-element bundle. This gas flow cools the aluminum boundary tube of the fuel-element bundle and isolates the calandria tube from the hot coolant flowing in the bundle. This low inlet temperature, plus an unusually large pumping power fraction, results in quite high specific power and power density, at the expense of rather low net efficiency (25.5 per cent). Some of the characteristics of the plant are summarized in Table III-2.

Table III-2 CHARACTERISTICS OF D_2O-MODERATED GAS-COOLED REACTOR FOR CZECHOSLOVAKIA[7,8]

Fuel	Natural U metal rods, 4 mm diam. × 4.0 m long
Fuel jacket	Mg–1.5% Be, 0.45 mm thick
Fuel subassemblies	Cylindrical bundles of 150–200 fuel rods, with variable spacing to allow for neutron flux dip in bundle
Core diameter	14 ft
Coolant	CO_2 at 60 atm (880 psi)
Maximum jacket temp.	500°C (932°F)
Outlet gas temp.	425°C (798°F)
Inlet gas temp.	100°C (212°F)
Steam conditions	29 atm (426 psi), 400°C (752°F) } H.P. turbine
	2 atm (29 psi), 180°C (356°F) } L.P. turbine
Circulating blowers	6, 4.65 Mw each
Boilers	12
Δp (reactor)	9 atm (132 psi)
Δp (entire circuit)	12 atm (176 psi)
Pressure vessel	Low carbon steel, ~60 ft high, >14 ft in diam., wall thickness, 5.5 in.

For the long-term future in the United Kingdom,[9] a concept is being investigated which has long been considered, and which was the subject of some rather concentrated work in the earliest stages of the U. S. reactor program,[10] but which has never been carried to the experimental reactor stage — the completely "ceramic" reactor. The excellent neutron economy and high-temperature potentialities of this reactor type are well known. The major deterrent to the development of the reactor type has been the circumstance that no methods are known for preventing contamination of the coolant gas by fission products. The British are approaching the development with an appreciation of the difficulty of the problem and with the expressed expectation that it will not be solved quickly.

The characteristics of the concept are quite high coolant temperature (well above 1000°F); very high conversion ratio (approaching a breeder if not actually a breeder) and very long fuel life; and relatively high power density (of the order 10 kw/liter). The high temperature is to be attained through the use of an inert gas (helium) as coolant with a completely nonmetallic reactor structure. The structure may be beryllium oxide or graphite. With either material, and with enriched fuel, the neutron economy can be quite high provided the reactor is large enough to prevent high neutron leakage. Granted a high conversion ratio and long reactivity life, the long fuel life is to be attained by mixing the fuel with the moderator and thus diluting the radiation damage effects to the maximum possible degree. The same dilution makes possible a high density of heat transfer surface in the reactor and is expected to make the target value of power density possible with modest coolant pressures in the range of 10 atm. With the "homogeneous" reactor design which is visualized, any positive neutron temperature coefficient of reactivity associated with the use of plutonium in the reactor would be also a prompt temperature coefficient and would be most undesirable. For this reason and for the reason of better neutron economy, the reactor is visualized as operating on the uranium-thorium fuel cycle.

The experimental reactor which has been proposed as a first step toward this development (after the hot critical experiment currently

under construction) uses graphite as the moderator and structural material.

The fuel elements consist of graphite rods containing a dispersion of U^{235} and thorium oxide, sheathed with graphite. The rods are arranged in clusters of seven. With this fuel arrangement there is no conventional barrier to the escape of fission-product gases. To reduce the leakage of fission products, each fuel rod contains at its upper end an inlet vent containing a porous plug through which a bleed flow of coolant gas passes, and at the bottom each of the seven rods is vented into a common duct in the base of the fuel cluster. The bleed coolant system inside the fuel rod is maintained at slightly lower pressure than the main coolant.

Sixty-one of the seven-rod clusters make up the core which is roughly a 4-ft-diameter cylinder 4 ft long. The closely packed rods have surfaces shaped to define a number of passages through which the coolant passes. Around the assembled fuel elements is a side reflector consisting of graphite columns 8 ft high with an over-all diameter of 10.5 ft. The top and bottom reflectors are each 2 ft thick. The helium coolant enters and leaves the pressure vessel through ducts located just above the core. Entering coolant (660°F) flows downward around the outside of the graphite stack and subsequently flows upward through the core, emerging at a mixed mean temperature of 1380°F.

Control of the reactor is exercised by means of a curtain of 30 absorbing rods, moving in vertical channels near to the inside boundary of the reflector. These provide a small negative reactivity for shutdown, and their removal compensates for temperature coefficient, poison, and depletion, a total range of reactivity between 14 and 15 per cent being controlled. The possibility of using boron as a burnable poison in the fuel inserts as a means of reducing the depletion compensation required from the control rods is also considered. It is proposed that the control rods would be actuated by cable drives with the provision of magnetic clutches in the winding mechanisms to release the cables in the event of an emergency shutdown requirement. To protect the actuators from any primary circuit activity, zirconium sleeves are provided which isolate these control mechanisms from the circuit gas and thereby simplify maintenance.

In a reactor of this type it is important that one should be able to load and unload fuel elements without disturbing the helium coolant in the primary circuit. Accordingly, a charge/discharge machine capable of transferring elements through a gas lock is required. The relative infrequency of fuel charging would enable the work to be carried out at a convenient shutdown period when gas temperatures are low and pressure correspondingly reduced. During reactor operation the charge/discharge machine would be parked in a shielded position where the temperature is kept low by a

Table III-3 SUMMARY OF DESIGN CHARACTERISTICS FOR THE U. K. HIGH-TEMPERATURE GAS-COOLED REACTOR[9]

Reactor thermal power, Mw	10
Core length, ft	~4
Core diameter, ft	~4
Reflector thickness, ft	2.00 (axial) 3.25 (radial)
No. of fuel channels	61
Reactor coolant	Helium
Gas pressure at reactor inlet, psia	150
Gas temp. at reactor inlet, °F	660
Gas temp. at reactor outlet, °F	1380
Fuel material	UO_2/ThO_2 dispersion in graphite
Fuel loading, Kg U^{235}	14.2
Graphite/U^{235} atomic ratio	2860
Thorium/U^{235} atomic ratio	7.4
Mean fuel temp., °F	1880 (1300°K)
Mean graphite temp. in core, °F	1520 (1100°K)
Core void, %	15
Initial excess reactivity (10 Mw), %	5.9
Excess reactivity available in cold (300°K) clean condition, %	11.3
Reactivity absorption due to thorium temp. (300 to 1300°K), %	−2.0
Reactivity absorption due to moderator temp. (300 to 1100°K), %	−5.5
Reactivity released by heating reflector through mean temps. of 375°K (side) and 500°K (ends), %	+2.1
Reactivity associated with equilibrium Xe, %	2.5
Reactivity associated with equilibrium Sm, %	0.7
Maximum reactivity absorption by Xe, 6.5 hr after shutdown, %	4.0
No. of fuel assemblies	61
No. of fuel rods per assembly	7
Lattice type	Triangular
Lattice spacing, in.	~2.25
Fuel sheath material	Graphite
No. of control rods	30
Control-rod locations	Inside boundary of reflector
Reactor vessel diam., ft	~11
Reactor vessel over-all length, ft	~40
Vessel wall thickness, in.	~2

cooled by-pass of gas which subsequently flows back into the reactor.

The fission products pumped from the bottom of the fuel elements are passed through a fission-product trapping system based on the use of active charcoal absorption beds. The emergent helium will still contain the longest period krypton and xenon activities since these are merely delayed. This gas will also go through a chemical purifier to remove O_2, CO_2, CO, and N_2O before returning to the main cooling circuit.

The characteristics of the reactor are summarized in Table III-3.

Once the principle of the leaky fuel element has been accepted, a great variety of reactor designs is possible to exploit the advantages which are bought at the price of fission-product leakage. One of these was described in a proposed reactor design from a paper[11] from the Federal Republic of Germany presented at the Geneva Conference. This proposal is for a pebble bed reactor, in which the reactor consists of a pressure vessel filled with pebbles of graphite of diameter 5 to 6 cm. Fertile and fissionable material are contained inside the pebbles in the form of cores of uranium or thorium carbides. The pebbles are poured into the top of the reactor, and exposed pebbles are withdrawn as necessary through a funnel-shaped opening at the bottom. If the British high-temperature reactor experiment is built, it should yield valuable information not only for the proposed British concept but for all such allied concepts which have been proposed from time to time in the past.

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Moderator Properties of Water and Graphite

It has been pointed out (Section II) that the moderator properties of H_2O are not good enough to sustain a chain reaction with natural uranium, whereas those of graphite are, but with only a little reactivity to spare. Therefore one might expect that in a graphite-moderated water-cooled reactor nuclear properties approaching those of the graphite-moderated gas-cooled reactor might be attained only if the water fraction is kept quite low. That is, natural-uranium operation might be possible, and conversion ratios would be somewhat less than 0.85. However, it may be expected that if the H_2O to uranium ratios begin to approach those used in water-moderated reactors, the nuclear properties will also approach those characteristic of such reactors: the neutron absorption of H_2O will dominate the neutron economy, and the graphite will enter the neutron physics picture only as additional moderator which will reduce the resonance absorption of U^{238} but not enough to allow criticality with natural uranium.

That the above arguments are substantially correct is demonstrated in Fig. 9, which shows the results of buckling measurements[1] on graphite-moderated lattices containing several different proportions of H_2O. Since the buckling must be substantially above zero to allow criticality in a reactor of reasonable size, it is quite evident that a relatively small fraction of H_2O suffices to eliminate the possibility of natural-uranium operation. Nevertheless the tolerable H_2O content is high enough to make such a reactor feasible.

It is not necessarily true that the desire to achieve neutron physics performance characteristic of graphite moderation is the only reason for employing graphite as moderator in H_2O reactors. Graphite might be considered also as a "filler" material, one which can be used to fill in spaces between fuel elements without a reactor physics penalty (and usually with some improvement in the physics characteristics) and thus allow wide spacing of the fuel elements and the use of structural concepts for the reactor which are out of the question when H_2O alone is used as moderator.

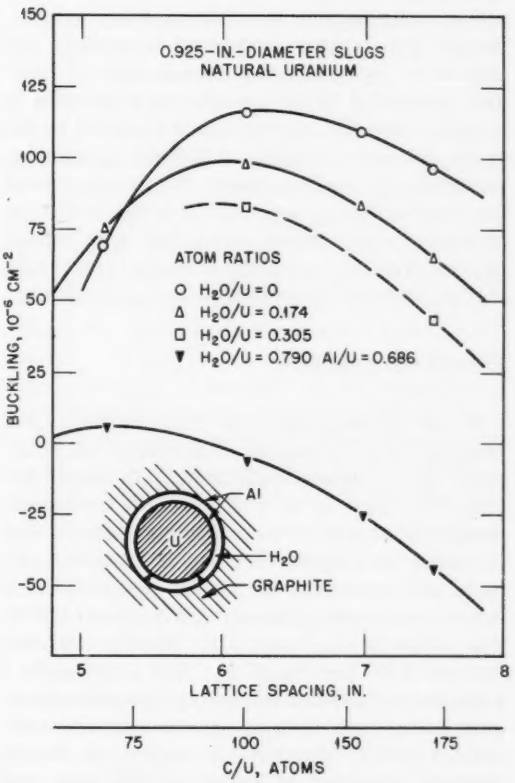

Figure 9—Measured bucklings of graphite–uranium metal lattices containing various fractions of H_2O. (Reproduced from Ref. 1.)

Problems of Structure and Design

To date the reactor designs for water-cooled graphite-moderated reactors have aimed at preserving as nearly as possible the neutron

physics characteristics of graphite moderation and have limited the ratio of water to fuel rather drastically. The heat removal capacity is then restricted by the small amount of water available for cooling in the coolant channels and by the practical impossibility of achieving a high ratio of heat-transfer area per unit mass of fuel and at the same time a low quantity of coolant per unit mass of fuel. Consequently, the specific power of such reactors may be expected to be lower by a considerable amount than that which is characteristic of reactors cooled and moderated by H_2O.

Obviously there is no attraction in a pressure-vessel type reactor moderated by graphite and cooled by H_2O since the combination of large size demanded by the moderating properties of graphite and the high pressure required by the vapor pressure relation of H_2O set impractical requirements for the vessel. On the other hand the combination is well suited to the use of the pressure-tube concept since the wide lattice spacing typical of graphite reactors is favorable for the mechanical arrangement of such a design.

Hanford Studies

In the United States, the Hanford Works has been the most consistent proponent of the pressure tube, graphite-moderated, H_2O-cooled design. They have in the past prepared conceptual designs of at least two such power reactors, one operating on slightly enriched uranium (0.8 per cent) and employing zirconium pressure tubes to produce saturated steam[2] at 315 psi and 422°F. The other design[3] was also slightly enriched (0.8 to 0.85 per cent) and was a boiling and superheating reactor employing zirconium pressure tubes in the boiling section and stainless-steel tubes in superheater section; it was to produce superheated steam at 850 psig and 900°F.

USSR Siberian Station

During the Geneva Conference, the USSR announced that it has a graphite-moderated H_2O-cooled reactor of 100-Mw (electrical) capacity operating in Siberia. The reactor is said to be the initial unit of a station which will ultimately have an output of 600 Mw (electrical). The plant produces steam at rather low temperature and is fairly obviously a dual-purpose unit for the production of plutonium as well as power. It would hardly be attractive as a pure power producer, but it is nevertheless an impressive installation and one which gives the USSR claim to the highest rated power reactor now in existence.

The available information on the plant is meager, consisting of a very brief announcement of the reactor capabilities and the information contained in a motion picture exhibited at the Conference. Table IV-1 gives some of the possible characteristics, as inferred from these sources.

Table IV-1 POSSIBLE CHARACTERISTICS OF SIBERIAN REACTOR

Electrical output, Mw	100
Fuel	Natural U (metal)
Moderator	Graphite
Coolant	H_2O
Pressure tubes and fuel jackets	Al-Si alloy
Core diam., ft	35*
Core length, ft	25*
No. pressure tubes	2300*
Fuel loading, tons	250*
Lattice spacing (square), in.	7.7*
Conversion ratio	0.75*
Coolant inlet temp., °F	356
Coolant exit temp., °F	428
Steam temp. (superheated), °F	365
Steam pressure, psi	<163
No. turbines	3

* These numbers are crude estimates only.

The reactor is graphite-moderated and water-cooled and operates on natural-uranium metal. Apparently it uses an aluminum-silicon alloy for the pressure tubes and fuel jackets. The reactor is mounted vertically above a large water pool into which spent fuel elements are discharged. The pressure tubes apparently are of the bayonet type with both inlet and outlet connections at the top. The reactor appears to be loaded from the top and unloaded from the bottom, but it is not clear whether the elements are discharged from the pressure tube or whether the pressure tube and fuel are discharged together as a replaceable unit. The entire reactor and shield lie below the floor level of the reactor room; the control-rod drives, which are of the cable and drum type, are mounted at floor level.

USSR Superheating Water-cooled Reactor

Of much greater interest as a power reactor is a second graphite-H_2O reactor, presently under construction in the USSR,[4] which is to produce directly superheated steam at about 930°F and 1620 psi. This reactor, 100-Mw electrical output, is to be the first unit of a station ultimately intended for a capacity of 400 Mw (electrical). The reactor is a direct descendant of the Russian First Atomic Power Station, retaining most of its characteristic features, but improving the neutron physics by the use of more massive fuel elements (and of course much larger reactor size), and adding the feature of nuclear superheat. The main characteristics are given in Table IV-2.

Table IV-2 CHARACTERISTICS OF USSR SUPERHEATING WATER-COOLED REACTOR

Power, thermal, Mw		285
Power, electrical (gross), Mw		100
Thermal efficiency (gross), %		35.1
Core diameter, ft		23.6
Core height, ft		19.7
Reflector thickness (graphite), ft		2.6
Lattice spacing (square), in.		7.9
Graphite coolant		Nitrogen
No. boiling fuel assemblies		730
No. superheating fuel assemblies		268
No. control and safety rods		100
No. steam turbines		1
Turbine capacity, kw		100,000
Turbine inlet pressure, psi		1320
Turbine inlet temp., °F		930
Average fuel-cycle time, days		730
U charge, tons (metric)		90
U enrichment at beginning of cycle, %		1.3
U enrichment at end of cycle, %		1.03
Conversion ratio at beginning of cycle		0.65
Conversion ratio at end of cycle		0.55
U^{235} burned during cycle, kg		243
U^{235} fissioned during cycle, kg		198
Pu^{239} burned during cycle, kg		55
Pu^{239} fissioned during cycle, kg		34
Pu^{239} plus Pu^{241} remaining at end of cycle, kg		132
Excess reactivity for temp. coefficient		0.040
Excess reactivity for Xe and Sm		0.015
Excess reactivity for burn-up and long-lived fission-product poisons		0.025
Total excess reactivity		0.080
For hottest fuel assembly	**Boiling**	**Superheating**
Heat output, kw	405	363
Heat flux, Btu/(sq ft)(hr)	194,000	177,000
Max. coolant velocity, ft/sec	32.8	187
Max. temp. of pressure tube, °F	670	987
Max. U temp., °F	752	1022
Max. graphite temp., °F	1220	1340

The reactor, like that of the first atomic power station, is installed vertically and is loaded and unloaded from above. The basic distinctive feature of the reactor is the fuel subassembly. As embodied in the superheating reactor, the subassembly consists of a cluster of seven tubes, equal to the reactor core in length, embedded in a graphite cylinder. The central tube is a simple stainless-steel pressure tube which is fed with inlet water from an inlet connection at the top and conducts this water to the bottom of the subassembly where it is distributed to the six surrounding tubes, the fuel elements. Each fuel element is made up of a central stainless-steel pressure tube, inside diameter 9 mm and thickness 0.4 mm, which is surrounded by a fuel matrix consisting of a dispersion of uranium metal particles in magnesium (12 wt.% magnesium). This fuel annulus is in turn surrounded by a thin stainless-steel can 0.2 mm thick. There is no metallic bond between the fuel and the pressure tube; contact for heat transfer is a mechanical one, probably ensured by the stretching of the internal pressure tube when it is pressurized. Thus the subassembly is of the bayonet type, having both inlet and outlet water connections at the top. An approximate cross-sectional diagram of the subassembly is given in Fig. 10.

From the point of view of reactor physics, the geometrical arrangement of the fuel is good. The H_2O volume is reasonably low; the volume ratio of H_2O to equivalent full-density uranium is 0.75. Furthermore, the geometrical arrangement, with the water and its pressure tube inside the fuel annulus, minimizes the volume of high-cross-section stainless steel required and places both the water and the stainless steel in the region of minimum thermal neutron flux. As a result of this arrangement, and a lattice spacing which gives relatively high resonance escape probability (and relatively low conversion ratio), the reactor is expected to operate on uranium of only 1.3 per cent enrichment despite the use of stainless steel.

The same characteristics which give good physics performance (low water content, low ratio of pressure tube surface to fuel volume) penalize the heat-removal performance and the specific power. The specific power is relatively low compared to that usually achieved with

water cooling but is quite respectable in comparison with the usual values for gas-cooled reactors. The value is approximately 3 Mw thermal per ton of uranium metal. At U. S. prices for enriched uranium, the use charge on the fuel in the reactor would amount to about 0.6 mil/kw-hr).

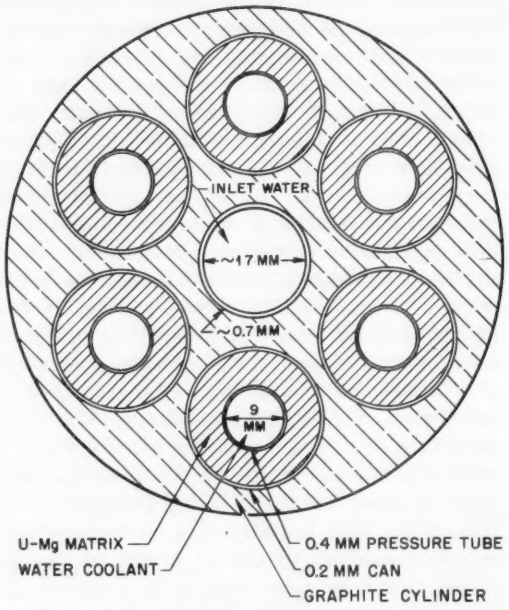

Figure 10—Approximate cross-sectional diagram of fuel assembly for superheating reactor. Diameters of fuel and graphite cylinders and spacing of elements are estimates only.

To achieve superheat, some of the fuel elements are designed to be cooled by steam instead of by liquid water. These elements are identical to the water-cooled elements except that a different grade of stainless steel is used for the pressure tube. The coolant–steam circuit is arranged as follows. Water is circulated in the liquid-cooled element in the usual way; these channels operate with net boiling, at a pressure of about 2300 psi, an entrance temperature of 572°F, and an exit temperature of 644°F. The steam–water mixture issuing from the elements goes to a steam separator, and the separated steam and water are sent to two separate heat exchangers, where they preheat and evaporate secondary water, giving secondary saturated steam at 1620 psia and 607°F. The secondary steam is sent directly through the superheating elements where it is heated to about 950°F.

Initially, on start-up, the superheating tubes are filled with liquid H_2O, which is later blown out by secondary steam as part of the start-up procedure, with the reactor operating at about 30 per cent full power. These elements are installed in the reactor at radial positions where the reactivity is least sensitive to changes in H_2O content; this is in an annulus intermediate between the center and edge of the reactor.

Many features of the reactor concept, including operation of prototype elements under boiling and superheating conditions, and the transition from water cooling to steam cooling, have been checked out on the first atomic power station.[5]

If the reactor performs according to design, it will be a most promising type of new reactor. It does have certain shortcomings, the most obvious of which is the rather short fuel life (2.7 kg per ton burn-up = 2300 Mwd/ton) claimed. The limitation on life is evidently expected to be a metallurgical one, and the desirability of attaining longer life is discussed in the paper describing the reactor. A fuel element limited to a burn-up of 0.27 per cent is of little interest in the United States economic picture. However, much higher burn-ups were reported on test fuel elements of the atomic power station type and could presumably be attained but perhaps at the cost of alloying materials which would increase the enrichment requirement substantially. At the present time too little is known about the reactor to allow evaluation of such specific deficiencies. The reactor is more interesting for its broader implications concerning the general concept of the tube type reactor and the direct nuclear superheating of steam.

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